UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated February 14, 1997, as supplemented by letters dated June 20, August 5, September 22, November 19, December 9, December 17, and December 31, 1997, January 23, February 12, February 26, March 3, March 6, March 16, April 3, April 13, and two letters on April 17, 1998, the Southern Nuclear Operating Company, Inc. (SNC) et al., submitted a request for changes to the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, Technical Specifications (TS) to increase the maximum reactor core power level for facility operation from 2652 megawatts-thermal (MWt) to 2775 MWt. The amendments also approve changes to the TS to implement uprated power operation. The results of the uprate evaluations and analyses were documented in Westinghouse WCAP-14723, "Farley Nuclear Plant Units 1 and 2 Power Uprate Project NSSS Licensing Report," dated January 1997 and submitted by SNC with the February 14, 1997, request.

The November 19, December 9, December 17, and December 31, 1997, January 23, February 12, February 26, March 3, March 6, March 16, April 3, April 13, and two letters on April 17, 1998, provided clarifying information that did not change the February 14, 1997, application and the initial proposed no significant hazards consideration determination (October 8, 1997, 62 FR 52588).

2.0 BACKGROUND

FNP Units 1 and 2 are currently licensed for operation at a reactor core power level of 2652 MWt. SNC undertook a program to uprate the FNP units to a maximum reactor core power level of 2775 MWt, approximately a 4.6 percent increase. At the core uprate power, the generator electrical output for each unit will increase approximately 25 megawatts-electrical (MWe). The engineering studies supporting the core uprate have been performed in accordance with guidance contained in Westinghouse Topical Report WCAP-10263, entitled "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated January 1983. This WCAP methodology, although not formally reviewed and approved by the NRC, was followed by North Anna, Salem, Indian Point 2, Callaway, Vogtle, and Turkey Point for their core power uprate and these uprates were found acceptable.

SNC's letter of February 14, 1997, submitted Westinghouse WCAP-14723, "Farley Nuclear Plant Units 1 and 2 Power Uprate Project NSSS Licensing Report," dated January 1997, which provided supporting documentation and analyses for the proposed changes. SNC stated that the results of the Nuclear Steam Supply System (NSSS) analyses and evaluations demonstrate that the FNP Units 1 and 2 NSSS is in compliance with applicable licensing criteria and requirements and can operate acceptably at the power uprate conditions.

SNC addressed the overall risk associated with the increase in rated thermal power and concluded that there is no impact on the calculated core damage frequency. The conclusion was based on the fact that all of the Final Safety Analysis Report (FSAR) transients were analyzed or evaluated at the uprated conditions with acceptable results and because SNC evaluated the success criteria and operator action failure probabilities used in the current FNP probabilistic risk assessment (PRA) model with no adverse impact.

The FNP power uprate amendments were reviewed with consideration given to the recommendations from the Report of the Maine Yankee Lessons Learned Task Group, dated December 5, 1996. This report is documented in SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997. The Task Group concluded that a power uprate review procedure should be developed in light of the Maine Yankee findings. Although a Maine Yankee lessons learned power uprate procedure has not been developed, the recommendations of the report were considered in the review of the FNP uprate. The main findings centered around the use and applicability of the code methodologies used to support the uprated power. SNC has made an effort to verify that the code inputs and assumptions are appropriate and applicable to the plant given the uprated conditions and any changes (plant modifications and procedural changes) that have occurred since initial licensing. SNC indicated that all principal codes were used in accordance with the applicable limitations and restrictions. SNC has also provided a summary list of analysis assumptions and of codes used and their review status (i.e., generically approved and new for FNP, generically approved and approved for FNP) for consideration in the NRC review. The staff considered all of the Maine Yankee Lessons Learned recommendations and appropriately addressed them in this review. For the few recommendations that were not adopted, the staff provided adequate justification and obtained cognizant management approval.

3.0 ACCIDENT ANALYSES EVALUATION

In support of this power uprate, the FNP units were reevaluated by SNC for operation at a rated thermal power of 2775 MWt with respect to safety analyses.

The transient analyses presented rely heavily on analysis performed in the past to support other NRC-approved licensing actions (overpressure delta temperature (OPΔT) - overtemperature delta temperature (OTΔT) review, VANTAGE-5 fuel conversion, and steam generator level tap relocation). SNC stated that these analyses were also used in accordance with the applicable code limitations and restrictions. SNC stated that the principal codes and methodologies used are all part of the FNP design basis with the exception of the methodology used to evaluate the large break loss-of-coolant accident (LOCA). SNC is using the NRC-approved WCAP-12945-P-A, "Code Qualification Document for Best Estimate LOCA Analysis," for the analysis of this accident. This methodology received a rigorous NRC review and approval. Sensitivity studies and comparative analysis were presented to the staff as part of the review effort. As a result, the staff concludes that a comparative analyses, between the old and new power levels, is not necessary, as recommended in the Report of the Maine Yankee Lessons Learned Task Group.

3.1 LOCA Analyses

3.1.1 Large Break LOCA (LBLOCA)

The LBLOCA analysis was performed, at the uprated conditions, using the NRC-approved Westinghouse Best Estimate Methodology or WCOBRA/TRAC. This methodology is appropriate for use at FNP because Units 1 and 2 are 3 loop Westinghouse designs for which the topical was approved. SNC performed the analysis in accordance with the code limitations and restrictions and as a result, the methodology is acceptable for use in FNP licensing applications, including reference in the technical specifications and core operating limits report (COLR). Use of this methodology is for the time from event initiation to core quench and not for use in evaluating long-term cooling.

The results of the best estimate analysis indicate that the calculated peak clad temperature (PCT) is 2064°F, the maximum localized oxidation is 12 percent, the maximum hydrogen generation is 0.6 percent, the core remains coolable, and the core remains cool in the long term. These results are acceptable and meet the acceptance criteria of 10 CFR 50.46 of a PCT less than 2200°F, a maximum localized oxidation less than 17 percent, the maximum hydrogen generation less than 1.0 percent, the core remains coolable, and the core remains cool in the long term. As a result, the staff finds this acceptable.

3.1.2 Small Break LOCA (SBLOCA)

SNC stated that the NRC-approved NOTRUMP continues to be used for the SBLOCA analysis, at the uprated conditions. In order to determine the limiting set of conditions, a number of cases were evaluated. The analysis was run at both the high and low average temperatures limits, the Units 1 and 2 flow characteristics (Unit 1 has a barrel/baffle upflow configuration while Unit 2 has a downflow configuration), and for both ZIRLO and zircaloy clad fuel. SNC stated that the analysis assumes the limiting single failure of a loss of one train of the emergency core cooling system (ECCS) with an assumed loss of offsite power (LOOP) at the time of the reactor trip. SNC stated that the delay from the time of event initiation to the time of the trip in the modeling is inconsequential and is not less limiting than assuming the LOOP occurs at the time of the event. The results indicate that the limiting PCT is 1968° F for the Unit 1, low T_{ave}, ZIRLO clad 3-inch break. The highest localized oxidation is 5.84 percent, the total oxidation remains less than 1 percent, the core remains coolable, and long-term cooling is maintained. As a result, the analysis of the SBLOCA is acceptable.

3.1.3 Hot Leg Switchover

SNC stated that a calculation has been performed to determine the new hot leg switchover (HLSO) time and minimum hot leg recalculation flow based on an uprated core power of 2775 MWt. The new HLSO time is 7.5 hours. The new hot leg recalculation minimum flow for the worst break and single failure is 89.1 lbm/sec. This hot leg recalculation minimum flow has been shown to be available. SNC concluded that with the above HLSO time and flow rate, the core geometry will remain coolable.

3.1.4 Post-LOCA Long-Term Cooling

SNC stated that an analysis of long-term cooling was performed at the uprated power. SNC has performed an evaluation to determine the effects of power uprating to post-LOCA long-term cooling. It is concluded that the new average temperature range has a very small effect on the post-LOCA sump boron concentration. Therefore, the core will remain subcritical post-LOCA and that decay heat can be removed for the extended period of time required by the long lived radioactivity remaining. The revised post-LOCA long-term core cooling boron limit curve is used to qualify the fuel on a cycle-by-cycle basis during the fuel reload process.

3.2 Non-LOCA Transient Analyses

SNC stated that the non-LOCA analysis was performed using codes that have been approved both generically and for FNP. They were used in accordance with applicable limitations and restrictions. A number of other NRC-approved license amendments were referenced because the previously approved analysis assumed uprated conditions. These include the transition to VANTAGE-5 fuel, the revision of the OPΔT and OTΔT setpoints with the implementation of the relaxed axial offset control, and the steam generator level tap relocation license amendments. SNC stated that these analyses were also performed in accordance with the applicable code limitations and restrictions. The analysis associated with the revision of the OPΔT and OTΔT setpoints included all of the changes associated with this amendment. The analysis associated with the VANTAGE-5 fuel transition included the uprated power, the ZIRLO fuel, and the reduced reactor coolant system (RCS) flow; however, it did not include all other conforming uprated related changes, including the reduction in ECCS flow. The transients that continue to rely on these analyses have been evaluated to verify continued acceptability. The staff finds this acceptable.

SNC stated that where applicable, the NRC-approved revised thermal design procedure (RTDP) was used in the evaluation of the departure from nucleate boiling (DNB). A reduced core flow of 86,000 gpm per loop was employed in the analysis, which is associated with an indicated flow limit of 97,800 gpm and 20 percent steam generator (SG) tube plugging was assumed. SNC stated that for each transient, both a stuck rod and the limiting single failure were assumed. The limiting single failure for each transient was selected on an event-specific basis. For the SG tube rupture, however, the single failure of the steam driven auxiliary feedwater pump was chosen, even though this is not the most limiting single failure. SNC stated that its design basis does not require the consideration of a single failure in the analysis of the SG tube rupture and the original analysis did not assume a single failure. The consideration of this single failure is more conservative than was required by the original design basis. As a result, the consideration of the failure of the steam driven auxiliary feedwater pump, a less limiting single failure, is now acceptable. Additionally, SNC stated that where an LOOP was assumed, the timing was chosen consistently with the original design basis assumptions. These assumptions are acceptable. A brief discussion of each individual transient is presented below.

3.2.1 Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical Condition

The uncontrolled RCCA bank withdrawal from a subcritical condition is analyzed to ensure that the core and the RCS are not adversely affected. This analysis was performed with acceptable results at the uprated conditions during the VANTAGE-5 fuel conversion. This has been demonstrated because the results of the analysis show that the minimum DNB ratio (DNBR) remains greater than the safety analysis limit and that the maximum fuel temperatures predicted to occur are much less than those required for clad damage or fuel melting to occur. The effect of the power uprate on this event is therefore, acceptable.

3.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

The analysis of the uncontrolled RCCA bank withdrawal at power case was analyzed while revising the OPΔT and OTΔT setpoints, at the uprated conditions, with acceptable results and NRC approval as addressed in Farley License Amendment No. 121 for Unit No. 1 and License Amendment No. 113 for Unit No. 2, issued on September 3, 1996.

3.2.3 Rod Cluster Control Assembly Misalignment

The dropped RCCA and the statistically misaligned assembly were analyzed at the uprated conditions while transitioning to VANTAGE-5 fuel. The analysis resulted in the calculated DNB meeting the design basis. Although the rod control system has been modified, affecting the control system response, SNC performed an analysis to demonstrate the acceptability of the rod control parameters since this modification. These calculations confirmed that the DNB design basis continues to be met. As a result, the staff finds this acceptable.

3.2.4 Uncontrolled Boron Dilution

The uncontrolled boron dilution was analyzed while revising the OPΔT and OTΔT setpoints, at the uprated conditions, with acceptable results and NRC approval as addressed in Farley License Amendment No. 121 for Unit No. 1 and License Amendment No. 113 for Unit No. 2, issued on September 3, 1996.

3.2.5 Partial Loss of Forced Reactor Coolant System Flow

The transient was analyzed for the power uprate. The event was analyzed using the RTDP and concluded that the DNBR design basis is met, the pressure of the primary and secondary remain within the design limits, and fuel centerline melt is not predicted. As a result, the staff finds this acceptable.

3.2.6 Startup of an Inactive Reactor Coolant Loop

The FNP TSs do not allow operation with a loop out of service, therefore this is not analyzed and it is acceptable.

3.2.7 Loss of External Electrical Load and/or Turbine Trip

The loss of load was evaluated for the uprate. The analysis was performed for two cases. The analysis was performed using minimum reactivity feedback with pressurizer pressure control (pressurizer sprays and power-operated relief valves (PORVs)) available using the RTDP to evaluate the DNB transient. The assumptions are established to maximize the DNB transient. The event was also analyzed using minimum reactivity feedback with no credit of pressurizer pressure control. These assumptions were made to maximize the primary and secondary overpressure transients. For the DNB transient analysis, the primary and secondary code safety relief valves are assumed to open at the low end of the allowable tolerance to maximize the DNB transient, and for the overpressure transient analysis, the valve is assumed to open at the upper tolerance. No accumulation in the primary safety relief valves is assumed; however, a delay of 1.6 seconds after the setpoint plus tolerance is reached is assumed prior to the valve relieving any steam to clear the loop seals. For the five secondary safety relief valves on each of three steam headers, the first three are modeled with 3 percent accumulation, the fourth 2 percent, and 10 psi accumulation for the last.

The analysis results conclude that for the DNB case, the DNB design basis is met. For the overpressure case, both the primary and secondary system pressure remains below 110 percent of the design pressure. The peak calculated pressure for the primary system is 2747 with a limit of 2748.5 psi. The calculated peak secondary pressure is 1199 psi with a limit of 1208.5 psi. Fuel centerline melt is not expected. As a result, the staff finds the analysis acceptable.

3.2.8 Loss of Normal Feedwater

The loss of normal feedwater was analyzed for the power uprate. The DNB transient for this event is bounded by the loss of electric load event. The peak pressure analysis results demonstrated that the primary and secondary systems do not exceed the safety limits. The pressurizer is not predicted to go water solid. Additionally, the motor driven auxiliary feedwater pumps to two SGs provide sufficient cooling. As a result, this is acceptable.

3.2.9 Loss of All Offsite Power to the Plant Auxiliaries

The loss of all offsite power to plant auxiliaries event was analyzed for the power uprate. The DNB transient for this event is bounded by the loss of electric load event. The peak pressure analysis results demonstrated that the primary and secondary systems do not exceed the safety limits. The pressurizer is not predicted to go water solid. As a result, this is acceptable.

3.2.10 Excessive Heat Removal Due to Feedwater System Malfunction

This event was analyzed for the VANTAGE-5 transition with acceptable results and NRC approval as addressed in Farley License Amendment No. 91 for Unit No. 1 and License Amendment No. 84 for Unit No. 2, issued on December 30, 1991. As a result, the current FSAR analysis remains valid.

3.2.11 Accidental Depressurization of RCS

The accidental depressurization of RCS was analyzed while revising the OPΔT and OTΔT setpoints, at the uprated conditions, with acceptable results and NRC approval as addressed in Farley License Amendment No. 121 for Unit No. 1 and License Amendment No. 113 for Unit No. 2, issued on September 10, 1996.

3.2.12 Inadvertent Operation of Emergency Core Cooling System During Power Operation

The inadvertent ECCS is analyzed to assure that the primary and secondary pressure limits are not exceeded, that the DNBR limits are not exceeded, and that the event does not progress to a more severe event. The event was analyzed and meets this criteria; however, to prevent the water solid relief from the primary code safety relief valves and the potential for the valve sticking open creating a more severe transient, SNC credits the opening of the PORVs. The staff has reviewed this and determined that this is acceptable because there is sufficient time for the operators to take action and open the PORV block valve if it is closed. Additionally, although the automatic actuation of the PORV is not considered safety-related, the accumulation circuits are routed separately; there are two separate Class 1E procured transmitters, powered from 1E power supplies, with 1E relays. Therefore, the PORV is considered highly reliable and its use is acceptable.

3.2.13 Inadvertent Loading of a Fuel Assembly into an Improper Position

SNC performed an analysis to verify that operation at the uprated power does not affect the ability of the instrumentation to detect the incorrect loading of an assembly. The evaluation concluded that the current analysis is still applicable and the conclusions remain valid. If a pin or rod were to be incorrectly loaded, the damage would be limited to that pin, if an assembly were incorrectly loaded, it would be detected by the incore detectors. The staff finds this acceptable.

3.2.14 Complete Loss of Forced Reactor Coolant Flow

The complete loss of flow was analyzed for the power uprate using the RTDP. For the limiting case run, the reactor is assumed to trip on the loss of RCS flow. The results indicate that the DNBR does not decrease below the limit, fuel centerline melt is not predicted, and the primary and secondary pressure also remain below the limit. As a result, the acceptance criteria are met.

3.2.15 Single Rod Cluster Control Assembly Withdrawal at Full Power

The single rod withdraw transient was analyzed during the VANTAGE-5 fuel conversion with NRC approval as addressed in Farley License Amendment No. 91 for Unit No. 1 and License Amendment No. 84 for Unit No. 2, issued on December 30, 1991. An evaluation was performed to confirm this analysis continues to be conservative with respect all the uprated conditions. The staff finds this acceptable.

3.2.16 Excessive Load Increase Incident, Accidental Depressurization of Main Steam System, Minor Secondary System Pipe Breaks, and Rupture of a Main Steamline

The main steamline rupture is considered a limiting fault; however, it was analyzed to more limiting criteria so that more frequent events do not need to be analyzed. The staff finds this acceptable. As a result, no DNB or fuel failures are predicted.

The event was analyzed using the NRC-approved Topical Report, WCAP-9226-P-A, Revision 1, "Reactor Core Response To Excessive Secondary Steam Releases," which is applicable to FNP. The excessive cooldown associated with the steamline rupture and a negative moderator temperature coefficient causes an increase in the core reactivity. The positive reactivity associated with the cooldown, and assuming the most reactive rod is stuck in the fully withdrawn position causes a return to criticality. The limiting case for the plant is in hot standby, end-of-life, 0 percent tube plugging, and minimum ECCS conditions. Both cases where offsite power is lost and offsite power is not lost were analyzed to determine which is more limiting. The case where power remains available is more limiting and no DNB and no fuel failure was predicted. As a result, it is appropriate to use this event to bound the more frequent excessive load increase and accidental depressurization of main steam events. The staff finds the results acceptable.

3.2.17 Rupture of a Main Feedwater Pipe

The feedwater pipe rupture was analyzed at the uprated power to support the relocation of the SG level tap license amendment. Although the analysis was performed at the uprated power, a sensitivity study was performed to quantify the effects of some of the other plant parameters being modified with this amendment. The sensitivity study concluded that there was no significant change to the peak primary and secondary pressure analysis and that there continues to be margin with regard to bulk boiling in the core and DNB. The auxiliary feedwater continues to provide sufficient capacity to remove the decay heat. As a result, the current FSAR analysis remains valid.

3.2.18 Single Reactor Coolant Pump Locked Rotor

The event was analyzed for the power uprate amendment. The results concluded that the peak primary and secondary pressure remain within the design limit of 110 percent of the design pressure. Although DNB occurs, less than 20 percent of the core is calculated to experience DNB and the maximum clad temperature was 2165 °F. As a result, the staff finds the analysis and results acceptable.

3.2.19 Rupture of a Control Rod Drive Mechanism Housing [Rod Cluster Control Assembly Ejection]

The rod ejection accident was analyzed based on the NRC-approved Topical Report WCAP-7588, Revision 1A, "An Evaluation of the Rod Ejection Accident in Westinghouse

Pressurized Water Reactors Using Spatial Kinetics Methods." The results conclude that the average fuel enthalpy at the hot spot remains well below 280 kcal/gm (actually below 225 kcal/gm) and therefore there is no danger of sudden fuel dispersal into the coolant. DNB is predicted to occur in less than 10 percent of the core and the peak RCS pressure does not exceed that which would cause reactor pressure vessel stress to exceed the faulted condition stress limits. As a result, the analysis is acceptable.

3.2.20 Steam System Piping Failure at Full Power

The steam system piping failure at full power was analyzed while revising the OPΔT and OTΔT setpoints, at the uprated conditions, with acceptable results and NRC approval as addressed in Farley License Amendment No. 121 for Unit No. 1 and License Amendment No. 113 for Unit No. 2, issued on September 3, 1996.

3.2.21 Steam Generator Tube Rupture (SGTR)

The SGTR analysis is used to verify that the postulated offsite dose consequences are acceptable. SNC performed mass and energy balance analyses, consistent with the original design basis, to determine the quantity of primary-to-secondary leakage from both the intact and faulted steam generators and the quantity of steam released to the environment.

3.3 Containment Integrity Analyses

SNC performed containment integrity analyses at uprated power to ensure that the maximum pressure inside the containment will remain below the containment building design pressure of 54 psig if a design bases LOCA or main steamline break (MSLB) inside containment should occur during plant operation. The analyses also established the pressure and temperature conditions for environmental qualification and operation of safety-related equipment located inside the containment. The LOCA peak pressure is also used as a basis for the containment leak rate test pressure to ensure that dose limits will be met in the event of a release of radioactive material to containment.

SNC indicated that the containment functional analyses included the assumption of the most limiting single active failure and the availability or unavailability of offsite power, depending on which resulted in the highest containment temperature and pressure. Bounding initial temperatures and pressures for analyses were selected to envelope the limiting conditions for operation.

3.3.1 LOCA Containment Integrity Analyses

SNC performed analyses to determine the containment pressure and temperature response during postulated LOCAs using mass and energy (M&E) releases, which incorporate revised design parameters corresponding to 2775 MWt with updated computer modeling. As in the current licensing basis FSAR, the postulated LOCA analyses were performed for the double ended hot leg (DEHL) guillotine break of reactor coolant pipe and the double ended pump suction (DEPS) break. It has been determined that the DEHL break results in the most limiting pressure during the blowdown phase and that the DEPS break yields the highest energy flow rates during the post-blowdown period.

SNC indicated that the M&E releases in the containment were calculated for power uprate using Westinghouse Topical Report WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983, which is applicable to FNP. In these analyses, the 1979 ANS 5.1 decay heat model with 2 a sigma uncertainty factor was used. Westinghouse Topical Report WCAP-8264-P-A, Revision 1, "Westinghouse Mass and Energy Release Data for Containment Design," August 1975 for M&E release calculations was used in the original/current design basis analyses. The M&E release analyses were presented in attachment 5 of WCAP-14723 of the February 14, 1997, submittal. The updated Westinghouse Topical Report WCAP-10325-P-A computer code with the same methodology and assumptions (except the FNP specific data) have been utilized and approved on many plant-specific dockets for Westinghouse pressurized water reactors such as Turkey Point, Catawba, McGuire, Sequoyah, Watts Bar, Surry Units 1 and 2, Millstone 3, Beaver Valley 2, and Indian Point 2. The staff finds the use of Topical Report WCAP-10325 for LOCA M&E release calculations acceptable for FNP.

SNC stated that the uprated containment pressure and temperature response analyses were performed using the GOTHIC computer code developed under an EPRI contract from the older NRC code, FATHOMS. GOTHIC was developed under a fully qualified quality assurance program and has undergone extensive peer review. The original containment temperature and pressure analyses were performed using the Copatta computer code. Later analyses were performed using the Compact code. Previous blowdown model data were used with the GOTHIC modeling and compared to the previous analyses with consistent results with differences on the order of 1 psi and 10°F at peak conditions. SNC explained the reasons for slight differences in pressure and temperature for FNP in a letter dated December 17, 1997. The staff has reviewed the December 17 letter and finds the reasons for slight differences acceptable. Therefore, based on the comparison with previous modeling with consistent results, the staff finds the use of the GOTHIC computer code for FNP containment temperature and pressure analyses acceptable.

SNC stated that uprate analyses made maximum use of the previous analyses for input assumptions. Other than changes in M&E release input due to power uprate and updated codes, other key model/inputs were: (a) residual heat removal (RHR) heat exchanger properties (heat transfer coefficient and component cooling water flow) were modified to represent actual plant design data; (b) containment cooler performance model was modified for degraded cooler service conditions (The model uses a single degraded cooler in service); (c) initial containment temperature assumption was 127°F, which corresponds to the containment design operating bulk average temperature plus margin; (d) initial containment pressure assumption was 14.7 psia because the containment mini-purge fans are normally running and maintaining pressure at essentially atmospheric. Additionally, limiting cases for LOCA and MSLB were analyzed at the TSs initial containment pressures of +3 psig and -1.5 psig for maximum pressure and temperature evaluations.

For the DEHL break, the FNP uprate analyses calculated a containment peak pressure of 43.0 psig and peak temperature of 263°F. For the DEPS breaks, the uprate analyses calculated a containment peak pressure of 41.5 psig and a peak temperature of 261°F. The uprate calculated LOCA peak pressure and temperature of 43.0 psig and 263°F remain below the containment design pressure of 54 psig and design temperature of 280°F. The proposed power uprate will not affect the Net Positive Suction Head (NPSH) requirements of ECCS pumps as the peak LOCA temperature of 263°F remains below the preuprate LOCA temperature of 268°F.

3.3.2 Main Steamline Break Containment Integrity Analysis

SNC performed analyses to determine the containment pressure and temperature responses during postulated steamline breaks inside containment for limiting conditions for operation at uprated power. The uprate analyses were evaluated for initial power levels of 102 percent, 70 percent, 30 percent, and 0 percent and spectrum of break sizes similar to that in the current FSAR. The MSLB mass and energy release and the pressure and temperature analyses have included the effects of various single failures. The MSLB M&E releases were calculated using the LOFTRAN computer code. The staff has found the use of this code acceptable. A discussion of LOFTRAN is provided in a staff Safety Evaluation issued on May 27, 1986.

Containment temperature and pressure responses were calculated using the GOTHIC computer code. As indicated earlier for LOCA containment analysis, the staff has found the use of the GOTHIC computer code acceptable for FNP.

For the MSLB, the uprate analyses calculated a peak containment pressure of 52.4 psig and a peak air temperature of 383°F. The peak containment pressure at uprated conditions remains below the containment design pressure of 54 psig. SNC indicated that peak air temperature, which is 5°F more than the preuprate temperature of 378°F, will be for a very short duration and that the containment structure temperature will remain below the containment design temperature of 280°F at all times. Also, updated calculated pressure and temperature curves for LOCA and MSLB cases will remain bounded by the curves used for equipment qualifications.

Based on the preceding discussion, the staff finds the proposed change for power uprate will not affect the containment integrity as the calculated peak containment pressure and temperature remains below the containment design pressure and temperature, and therefore, is acceptable.

3.3.3 Short-term Subcompartment Analysis

SNC indicated that an evaluation was conducted to determine the effect of a power uprate on the short-term LOCA-related M&E releases that support subcompartment analyses discussed in Chapter 6.2 of the FNP FSAR. In the FSAR, a double-ended circumferential rupture of the RCS cold leg forms the basis for the steam generator compartments; a 100 in^2 reactor vessel inlet break forms the basis for the reactor cavity region, and both spray line break and surge line break were considered for the pressurizer compartment. The magnitude of the pressure differential across the walls is a function of several parameters, which include the blowdown M&E release rates, the subcompartment volume, vent areas, and vent flow behavior. The blowdown M&E release rates are affected by the initial RCS temperature conditions. Any changes in RCS volume, steam generator liquid/steam mass and volume, and differences in units, such as upflow verses downflow, have no effect on releases because of the short duration of postulated accident. Any volumetric changes are small and have no impact on the subcompartment model. Therefore, the only change that needs to be addressed for this program is the decrease in the RCS coolant temperatures. Based upon the results of the evaluation, the current design basis LOCA-related M&E releases, including the spray line and the surge line releases, could increase by a factor of 1.18 due to RCS temperature effects.

SNC stated that the subcompartment analysis was performed for the pressurizer compartment with the increased releases of 18 percent due to RCS temperature effects. Results of the analysis showed that the subcompartment wall differential pressure and uplift forces remain bounded by the previous design basis analysis. FNP is approved for leak-before-break (LBB) as per WCAP-12825, "Technical Justification for Eliminating Large Primary Loop Rupture as a Structural Design Basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plants," January 1991. This means that the current breaks (a double-ended circumferential rupture of the reactor coolant cold leg break for the steam generator compartments, and a reactor vessel inlet break for the reactor cavity region) no longer have to be considered for the short-term effects. Since the RCS piping has been eliminated from consideration, the large branch nozzles must be considered for design verification. This includes the surge line, accumulator line, and the RHR line. These smaller breaks, which are outside the cavity region, would result in minimum asymmetric pressurization in the reactor cavity. Additionally, compared to the large RCS double-ended ruptures, the differential loadings are significantly reduced. The decrease in M&E releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets the increased releases associated with decreased RCS initial coolant temperature. Therefore, the current licensing basis subcompartment analyses that consider breaks in the primary loop RCS piping (steam generator subcompartment and reactor cavity region), therefore remain bounding.

Based on the preceding review, the staff concludes that the uprating is acceptable as the subcompartment pressure loading analysis from high-energy-line ruptures remain bounded by the current FSAR analysis.

3.4 Additional Design Basis and Programmatic Evaluations

3.4.1 Containment Post-LOCA Hydrogen Generation

SNC indicated that the effect of power uprate was reviewed for the Zirconium(Zr)-water reaction, corrosion of construction materials in the containment, radiolytic decomposition of core and sump solution modes of post-LOCA hydrogen production, and for the capability of the combustible gas control system to maintain acceptable hydrogen concentration inside the containment.

SNC stated that the hydrogen generation due to Zr-water reaction is not affected by the power uprate since total quantity of Zr (in the fuel cladding) remains unchanged. Also, the post-LOCA containment temperature profiles to determine the corrosion rate for aluminum and zinc for power uprate remains bounded by that used in previous design basis analyses, and therefore, hydrogen generation due to corrosion rates will not be affected by power uprate. The hydrogen generation post-LOCA due to radiolytic decomposition of core and sump solution will increase by approximately 5 percent proportional to the increase in reactor power level.

SNC indicated that the increase in hydrogen generation rate due to power uprate is determined to have a negligible effect on the post-accident hydrogen mixing system. The analysis performed for uprated power showed that with no recombiner in service, the hydrogen concentration will not exceed 4 percent by volume for 17 days following a LOCA. Placing a hydrogen recombiner in service prior to the 18th day following a LOCA will maintain containment hydrogen levels below the lower flammability limit of 4 percent. Based on the above review, the staff finds that the power uprate will not impact the post-LOCA hydrogen control system.

3.4.2 Compliance with 10 CFR Part 50, Appendix R

SNC stated that the power uprate evaluation did not identify changes to design or operating conditions that adversely impact the post-fire safe shutdown capability in accordance with Appendix R. SNC also stated that there were no physical plant configuration changes or combustible load changes. The staff finds this acceptable. The staff may review, during a future onsite inspection, the supporting information for SNC's letter of March 6, 1998.

3.4.3 Station Blackout (SBO)

SNC performed evaluations of the impact resulting from plant operations at the proposed uprated power level on system response and coping capabilities for SBO events. SNC stated that current design basis temperature profiles in areas housing SBO-required equipment remain bounded for an SBO in an uprated plant.

With this evaluation, the staff has also reviewed whether this power uprate would affect current General Design Criterion (GDC)-17 and SBO requirements. Although the power uprate resulted in a small electrical load increase of the reactor coolant pumps and charging pumps on non-Class1E 4160 V buses, SNC's assessment has not indicated that power uprate would affect bus loadings and voltages to such a degree that the electrical onsite or offsite power system configuration would need to be modified. Therefore, the staff finds that the power uprate continues to satisfy the current GDC-17 requirement. SNC has also reviewed the SBO coping analysis, which is a function of offsite power design, emergency power configuration, and emergency diesel generator target reliability per Regulatory Guide 1.155, "Station Blackout." SNC finds that none of the SBO coping criteria are affected by power uprate. On this basis, the staff concludes that power uprate at FNP would not affect current SBO coping duration and would continue to meet the GDC-17 requirements.

Based on its review and the experience gained from the review of power uprate applications for similar PWR plants, the staff finds that the impact on system response and coping capabilities for an SBO event resulting from plant operations at the proposed uprated power level is insignificant.

3.4.4 Safety-Related Motor-Operated Valves (MOV)

In its response to the staff's request for additional information dated January 23, 1998 , SNC stated that the FNP NSSS and Balance of Plant safety-related valves were demonstrated to be acceptable for the power uprate. This determination was confirmed by verifying that changes in system operating temperature and pressure were bounded by the requirements of the associated equipment specification. As a result, SNC concluded that the power uprate has no impact on FNP's Generic Letter 89-10 (Safety-Related "Motor-Operated Valve (MOV) Testing and Surveillance") program and that the increased thrust required to operate the MOVs due to expected differential pressure conditions is within the capabilities of the existing valve actuators. On the basis of its review, the staff concurs with SNC's conclusion that the power uprate will have no adverse effects on the safety-related valves and FNP's MOV program. The staff may review, in a future onsite inspection, the supporting information for SNC's letter of January 23, 1998.

3.5 Radiological Analysis

SNC performed reanalyses of a select number of the FNP FSAR Chapter 15 accidents. Such reanalyses were required because of SNC's proposal to increase core power and to change various operating parameters as part of the power uprate. These changes would alter the releases from these postulated accidents.

The accidents for which SNC performed such reanalyses and the NRC guideline dose limits for these accidents are as follows:

1. Loss of AC Power - 2.5 rem whole body, 30 rem thyroid
2. Waste Gas Decay Tank Rupture
   - 500 mrem whole body
3. Large Break LOCA
   - 25 rem whole body, 300 rem thyroid
4. Main Steamline Break
   - preexisting spike case - 25 rem whole body, 300 rem thyroid
   - accident-initiated spike case - 2.5 rem whole body, 30 rem thyroid
5. Steam Generator Tube Rupture
   - preexisting spike case - 25 rem whole body, 300 rem thyroid
   - accident-initiated spike case - 2.5 rem whole body, 30 rem thyroid
6. Locked Rotor
   - 2.5 rem whole body, 30 rem thyroid
7. Fuel Handling
   - 6 rem whole body, 75 rem thyroid
8. Rod Ejection
   - 6 rem whole body, 75 rem thyroid
9. Small Break LOCA
   - 25 rem whole body, 300 thyroid

Consistent with the guidance in the Standard Review Plan (SRP) for some accidents, the reanalysis was performed at a core power rating of 102 percent (2831 MWt) of the proposed power uprate value (2775 MWt).

SNC implemented alternate repair criteria (ARC) for the steam generator tubes. Because of this, in the event of an MSLB accident, there may be additional primary-to-secondary leakage over and above that which would otherwise occur. For FNP, the value for accident-induced leakage is presently at 23.8 gpm and the TS for normal operating leakage is 150 gpd/SG (0.1 gpm/SG). At the accident-induced leakage rate of 23.8 gpm, it became necessary to reduce the TS values for maximum instantaneous and the 48-hour values for dose equivalent ^{131}I to the current values of 9 μCi/g and 0.15 μCi/g, respectively. The staff anticipates that, upon replacement of the SGs, SNC will request that the TS values for dose equivalent ^{131}I in primary coolant be restored to their previous values of 60 μCi/g for the maximum instantaneous value and to 1 μCi/g for the 48-hour value. Therefore, with the exception of the MSLB, wherever possible, the staff performed its accident analyses involving the TS values with primary coolant levels of dose equivalent ^{131}I at the usual levels of 60 μCi/g and 1 μCi/g.

The staff reviewed SNC's calculations and performed confirmatory calculations of the doses associated with these accidents for individuals located offsite at the exclusion area boundary (EAB) and at the low population zone (LPZ) and onsite for the control room operators. The FNP control room is designed to isolate and the control room emergency ventilation system automatically starts on a containment isolation signal. If the control room receives a high radiation signal, the control room is automatically isolated but it requires manual action on the part of an operator to initiate operation of the control room emergency ventilation system. Based on these assumptions, the staff found the control room operator doses acceptable.

The following sections provide the staff's assessment of the potential consequences of the above postulated accidents.

3.5.1 Accidents Analyzed

3.5.1.1 Loss of ac Power

SNC indicated that a loss of ac power would not result in the release of radioactivity unless there was radioactivity in the primary coolant and a primary-to-secondary leak existed. The analysis performed by SNC assumed a reactor coolant activity level based upon 1 percent failed fuel for noble gases, the TS values for dose equivalent ^{131}I in primary coolant and a 150 gpd/SG primary-to-secondary leak rate.

The staff assessed the potential consequences of a loss of ac power. The staff's assumptions are presented in Table 3.5.1-1. Thyroid and whole body doses were calculated at the exclusion area boundry, low population zone, and in the control room. The thyroid, whole body and control room operator doses are presented in Tables 3.5.2-1 through 3.5.2-3, respectively. The doses were found acceptable.

3.5.1.2 Waste Gas Decay Tank Rupture

SNC reevaluated the consequences of the release of the contents of a waste gas decay tank as a result of the power uprate. SNC indicated that its evaluation "conforms to the guidelines of Regulatory Guide 1.24. The releases result in offsite doses [which] are a small fraction of the 10 CFR guidelines, which meets the acceptance criteria."

The staff did not perform such a reassessment of the consequences of this accident because FNP is limited by TS 3.11.2.6 as to the quantity of radioactivity that each tank may contain. This limit is 70,500 Ci of ^{133}Xe equivalent. Since SNC did not propose to change the limit for this TS; the previous TS amendment, which approved the 70,500 Ci, was still deemed to be limiting.

It should be noted that the guideline dose criterion for the curie contents of a waste gas decay tank is 500 mrem to the whole body. This was detailed in Section 5.6 of NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." The guideline is not a small fraction of Part 100 (2.5 rem to the whole body) as stated by SNC in its submittal.

3.5.1.3 Large Break LOCA (LBLOCA)

SNC calculated the potential consequences of a postulated LBLOCA to the control room operators and to individuals located offsite at the exclusion area boundry and low population zone. SNC postulated that the pathways for releases in the event of a LOCA were containment leakage and ECCS recirculation loop leakage discharged via the penetration room filter system. SNC also assumed that at the time that the LOCA occurred, the containment was being purged. The purge was assumed to be isolated within 6 seconds following the accident. As part of the effort to demonstrate the capability of the plant to meet Part 100 and GDC-19 doses, SNC also performed a calculation that assumed that it was necessary to purge the containment 18 days after the LOCA due to the H_2 recombiners being inoperable.

In SNC's analysis, it was assumed that the containment source term for elemental and particulate forms of iodine was reduced by sprays. In addition, it was assumed that the elemental form of iodine was also subject to removal via plateout. SNC assumed that the sprayed and unsprayed regions were mixed by the containment cooling fans at a rate of 40,500 cfm. However, the staff did not incorporate this assumption because the TS associated with the containment cooling fans places no requirements for fan flow. Consequently, there is no basis for assuming that mechanical mixing will occur at the above stated rate. Consequently, the staff assumed that the flow rate between the sprayed and unsprayed regions was due to natural convection and was equal to two of the unsprayed regions volume per hour.

SNC also assumed that the elemental forms of iodine were removed from the containment due to plateout. SNC assumed that the removal rate was a function of the elemental iodine, which had been removed. Until a decontamination factor (DF) of 100 was achieved, the removal rate was 2.7/hr. Between a DF of 100 and 1000, the removal rate was 0.27/hr. After a DF of 1000 was achieved, no removal was assumed to occur due to plateout. The staff reviewed this assumption and determined that it was acceptable.

WCAP-14723, Attachment 6, Tables 2.16-1 and 2.16-2 contain revised values for the post-accident iodine removal coefficients. Evaluation of these tables is presented below:

- In calculating the coefficient for plateout of elemental iodine activity released to containment (λ_w), SNC used wall deposition velocities for elemental iodine from NUREG/CR-0009. Using these deposition velocities, SNC calculated λ_w=12.5 hr $^{-1}$. However, NUREG-0009 specifically recommends that, in determining plateout coefficients, the mass transfer coefficient through gas film be used instead of wall deposition velocities. The same recommendation is made in Section 6.2.5, Revision 2 of the Standard Review Plan. Using this coefficient, the staff calculated λ_w=2.7 hr $^{-1}$. In its letter of March 3, 1998, SNC changed the coefficient for plateout of elemental iodine to a λ_w value of 2.7 hr $^{-1}$, which is an acceptable value.

- When the concentration of iodine in the containment atmosphere is reduced and the DF for containment iodine reaches certain limiting value, iodine removal becomes less efficient, and the calculated coefficient for the removal of particulate iodine by containment sprays (λ_p) should be reduced by a factor of 10. In Section 6.5.2, Revision 2 of the Standard Review Plan, this limit is specified as DF=50 and was utilized in the staff's calculation. However, the value of DF=100 as specified in NUREG/CR-0009 is a conservative number. In SNC's submittal, it is given as DF=100. The value of DF=100 will have an insignificant effect on radiation doses and therefore is acceptable.

The staff did note some inconsistency in the manner in which SNC stated that it had performed the calculation of releases from ECCS leakage and the actual manner in which it was performed. In Table 2 of a December 9, 1997, letter to the staff, SNC indicated that it had assumed that the leakage associated with ECCS operation would be 20 times the value in FSAR Table 6.3-8. However, calculations provided in a March 3, 1998, transmittal to the staff, showed that the value actually used was 10 times the rate in FSAR Table 6.3-8. The staff chose a value of 20 times the flow rate in FSAR Table 6.3-8 for this assessment.

SNC stated that the results of its analyses demonstrated that in all cases, Part 100 doses and GDC-19 doses were met.

Additional details on the assumptions for this evaluation are presented in Table 3.5.1.3-1. The staff assessed the potential consequences of a LOCA based upon the assumptions in this Table. The thyroid, whole body and control room operator doses are presented in Tables 3.5.2-1 through 3.5.2-3, respectively. The doses were found to be acceptable in that they were less than 25 rem whole body (or 300 rem to the thyroid) at the exclusion area boundary and control room doses were less than 5 rem whole body during an accident.

3.5.1.4 Main Steamline Break

FNP has applied the ARC to its SG tubes. Application of this criteria permits SNC to make the decision to allow tubes to remain in service when previously SNC would have been required to plug or sleeve the tubes due to the degradation that has occurred. However, allowing these tubes to remain in service necessitates that SNC take into account, in its analysis, the potential, which exists, that an MSLB may induce leakage in the tubes that have been allowed to remain in service due to application of ARC. Consequently, when an analysis is performed of the consequences of an MSLB accident concurrent with a loss of offsite power, SNC assumes that the accident induces leakage in those tubes of the affected SG which has experienced the pressures associated with a steamline break and to which ARC has been applied.

On October 29, 1997, Amendments 132 and 124 were issued to FNP Units 1 and 2, respectively, which implemented the most recent ARC criteria. This amendment approved a primary-to-secondary leak rate of 23.8 gpm and primary coolant activity levels of dose equivalent ^{131}I of 9 µCi/g for the maximum instantaneous value and 0.15 µCi/g for the 48-hour value. On March 16, 1998, SNC submitted a revised MSLB analysis that utilized the assumptions of Amendments 132 and 124 but modified the assumptions to incorporate changes associated with the power uprate such as steam released from the intact SGs.

For the power uprate amendment, the reevaluation of the MSLB involved two cases. One case assumed the accident occurred following an iodine spike, referred to as the preexisting spike case. The second case assumed that the MSLB resulted in the initiation of an iodine spike, referred to as the accident-initiated spike. In both cases, a 150 gpd/SG, primary-to-secondary leak was assumed for the intact SGs. For case one, it was assumed that a preexisting iodine spike had occurred prior to the steamline break. Reactor coolant concentration was assumed to be at the TS Figure 3.4-1 full power limit of 9 µCi/g of dose equivalent ^{131}I.

The second case assumed the steamline break initiated a concurrent iodine spike. The reactor coolant concentration was assumed to be at the existing TS 48-hour limit of 0.15 µCi/g dose equivalent ^{131}I. The secondary system activity was assumed to be at the TS limit of 0.1 µCi/g dose equivalent ^{131}I. Concurrent with the MSLB, an iodine spike was assumed to occur that releases iodine from the fuel gap to the reactor coolant at a rate that is 500 times the normal iodine release rate. No failed fuel was assumed to occur as a result of the MSLB.

For both analyses, it was assumed that the 23.8 gpm of primary-to-secondary tube leak occurred in the faulted SG until it was isolated. After isolation of the faulted SG, it was assumed that primary-to-secondary leakage to the intact SGs would continue at a rate of 150 gpd/SG. Because offsite power is assumed to be lost, the main condenser was unavailable for steam dump and cooling of the reactor core must occur through the use of the safety valves. After 8 hours, no further steam release or activity release was assumed to occur due to the steamline break.

The power uprate amendment request presented a new value for the quantity of steam released from the intact SGs during an MSLB. The quantity presented in the power uprate amendment was approximately 25 percent less than the value which was presented in support of the latest ARC amendments. This decrease in steaming rate for the intact SGs incorporated a plant-specific value for Farley whereas the previous value had been a generic Westinghouse value. A rigorous calculation of the dose consequences of an MSLB would include the contribution from the steaming of the intact SGs. However, the dose contribution from this source is minimal. The staff's analysis, which assessed the latest ARC amendment request, assumed, in the calculation of doses, that any primary-to-secondary leakage to the intact SGs was released to the environment without a partition factor. This is a conservative assumption.

Since the primary-to-secondary leakage rates remained unaffected by the power uprate, the doses, which were calculated by the staff, remain unchanged. Based upon the manner in which the staff calculated the doses for the MSLB in the ARC amendments, these doses remain relevant and acceptable and are incorporated into this safety evaluation.

SNC was also asked to determine if the MSLB remained the limiting accident with respect to radiological consequences when the effects of ARC and the power uprate are considered or if the locked rotor or the rod ejection accidents resulted in accident-induced leakage which, when coupled with the increase in source term, became more limiting. The staff's concern was that the pressure transient associated with the locked rotor and rod ejection accidents might induce leakage during the event. When this accident-induced leakage is combined with the normal operating primary-to-secondary leakage and the failed and melted fuel associated with the transient, the potential exists that one of these accidents may become more limiting, from a radiological standpoint, even though the partition factor for the faulted SG is 1 for the MSLB and 100 for the other transients, provided the tubes are covered. SNC indicated that for the power uprate, even with the inclusion of ARC, the MSLB is the more limiting accident. SNC stated that the basis for this statement was contained in WCAP-12871, Rev. 2, "J. M. Farley Units 1 and 2 Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates," February 1992. Section 11.3 of WCAP-12871 had concluded that the increased source terms associated with the locked rotor and rod ejection accidents were offset by reduced primary-to-secondary differential pressure; decreased flashing and increased mixing in the SG; and continued coverage of the SG tubes at the tube sheet and the tube support plate interfaces. The issue of partial SG tube uncovery was addressed generically by the Westinghouse Owners Group and documented in WCAP-13247, "Report of the Methodology for the Resolution of the Steam Generator Tube Uncover Issue," in March of 1992. SNC stated that the conclusions drawn in the study, that the current design basis methodologies remain valid, is still applicable to Farley given the power uprate and the implementation of ARC.

SNC also presented a qualitative assessment to demonstrate that the MSLB was still limiting, with respect to radiological consequences. An assessment of the expected primary-to-secondary differential pressure associated with the locked rotor and rod ejection accidents determined that the locked rotor pressure transient was more challenging with regard to primary to secondary differential pressure. For the locked rotor accident, the primary pressure peaks in approximately 3 seconds for Farley and levels off after about 25 seconds. Farley estimated the accident-induced leakage for a locked rotor event. This scoping assessment relied on leakage data obtained previously for five pulled tube specimens from the Farley SGs. Leakage data for these specimens was collected for differential pressure loadings representative of normal operating conditions and of MSLB. This data was adjusted to reflect the pressure, temperature, and hydraulic conditions associated with the postulated locked rotor event. The adjustment procedure was adapted from an adjustment methodology included and documented in the industry ARC database for 7/8-inch tubing. This data base has been accepted by the NRC as documented in GL 95-05 and plant-specific SERs approving implementation of the ARC. Application of the adjustment procedure yielded a factor of 3.7 reduction in leakage rate for the first 25 seconds of a postulated locked rotor event when compared to leakage under postulated MSLB conditions and a factor of 8.4 reduction after 25 seconds.

SNC's estimates are based on limited data that required significant adjustment to reflect postulated conditions during the locked rotor event. Nevertheless, SNC's estimates incorporate a number of conservatisms such that it is the staff's judgement that actual leakage under postulated locked rotor and rod ejection events would be less than that indicated by SNC's assessment. In particular, SNC took no credit for the constraint against leakage provided by the corrosion product in the tube-to-tube support plate (TSP) crevices. Such constraint has typically not been assumed in assessing leakage from MSLB due to potential axial displacement of the TSP during the event. TSP displacement would not be expected to occur during locked rotor and rod ejection events.

SNC presented the results of offsite dose assessments based on the leak rate ratios above and a limiting MSLB leak rate of 24 gpm. The locked rotor estimate for leakage was utilized for the rod ejection accident since its value bounded both assessments. For the locked rotor, accident-induced leakage was estimated at 6.5 gpm per SG for 0-25 seconds and 2.9 gpm per SG for the remainder of the accident. These leakage rates were assumed to exist for all three SGs and long-term leakage was assumed to last until the RCS and SG pressures equalize.

It was assumed for the locked rotor that the leakage remained constant until residual heat removal cut in at 8 hours after that accident. For this qualitative assessment, SNC determined that a fuel failure value of 6.3 percent is sufficiently conservative; 20 percent was assumed in the licensing basis calculations. The staff found the value of 6.3 percent acceptable. Based upon these assumptions, SNC calculated a thyroid dose of approximately 22 rem.

For the control rod ejection, the duration of the accident, as described in the Farley Nuclear Plant Units 1 and 2 Power Uprate Project NSSS Licensing Report (WCAP-14723) was 2500 seconds. Using the rod ejection source term for the power uprate with the above noted primary-to-secondary leak rates, SNC calculated at offsite thyroid dose of approximately 46 rem.

Based upon the results for the locked rotor and rod ejection accidents noted above, since the consequences of the MSLB is at the existing dose limit of 30 rem, SNC concluded that the MSLB remains the limiting accident. SNC concluded this because the MSLB is at 100 percent of the dose guideline value whereas the locked rotor and rod ejection accidents doses were determined to be 60-70 percent of their guideline values. SNC also concluded that the offsite consequences would be more limiting than the control room operator doses (GDC-19).

The staff has evaluated SNC's assessment of whether the MSLB remains the limiting accident. The staff reviewed SNC's discussion of the pressure transients associated with the locked rotor and rod ejection accident, SNC's estimate of the accident-induced leakage, which may be initiated by such events, the estimated fuel failures associated with the locked rotor and rod ejection events and the radiological consequences postulated for these events. The staff concluded that the postulated offsite doses by SNC were reasonable. In addition, the staff concluded that it would be the offsite consequences rather than GDC-19, which would be limiting for this assessment. Based upon its review, the staff has concluded that the MSLB remains the limiting accident, with respect to radiological consequences, even when both ARC and the power uprate are considered.

The details of the staff's assessment of the doses of an MSLB were presented in Amendments 132 and 124. The doses, which were presented in that assessment, are presented in Tables 3.5.2-1 through 3.5.2-3 for the sake of completeness.

3.5.1.5 Steam Generator Tube Rupture

SNC reevaluated the consequences of a postulated SGTR accident. For this evaluation, SNC assessed two cases. One case involved a preexisting spike. The other case involved an accident-initiated spike. In both cases, in the intact SGs, it was assumed that primary-to-secondary leakage existed at the present TS value of 150 gpd/SG prior to and following the SGTR.

In the preexisting iodine spike case it was assumed that the iodine spike occurred prior to the SGTR. At the time that SNC performed its assessment of this accident for the power uprate, the existing technical specification value for the maximum instantaneous value of dose equivalent ^{131}I in the reactor coolant was 30 μCi/g. It was at this value that SNC performed its dose assessment and submitted it to the staff for review and approval. Subsequently, in Unit 1 Amendment 132 and Unit 2 Amendment 124, SNC reduced this value of dose equivalent ^{131}I to 9 μCi/g.

For the accident-initiated spike case, it is assumed that the SGTR initiates a concurrent iodine spike. For this assessment SNC assumed that the reactor coolant was at the 48-hour TS for dose equivalent ^{131}I. As for the preexisting spike case, SNC performed the analysis at the SNC value (0.5 μCi/g) which was in existence at the time the power uprate amendment request was filed. The secondary system activity was assumed to be at the TS normal operation limit of 0.1 μCi/g of dose equivalent ^{131}I. Concurrent with the SGTR, an iodine spike was assumed to occur that results in the release of iodine from the fuel gap to the reactor coolant at a rate that is 500 times the iodine release rate associated with the 48-hour TS value. SNC's analysis of the SGTR concluded that this accident resulted in no failed fuel.

For both analyses, it was assumed that offsite power was lost and the main condenser was unavailable for steam dump. SNC assumed that it took 8 hours to cool the reactor down to a point where no further release of steam or radioactivity to the environment would occur.

The staff has performed its assessment of the potential consequences of an SGTR event. The staff's assessment assumed that the reactor coolant activity level for dose equivalent ^{131}I was at 60 μCi/g for the preexisting spike case and at 1 μCi/g for the accident-initiated spike case. The staff performed its assessment at this value because the staff anticipates that SNC will restore these values to the pre-ARC TS values upon replacement of the FNP SGs.

There were certain issues associated with SG operation that the staff believed needed to be addressed in connection with the power uprate. These issues were SG overfill, SG flashing fractions, and tube uncovery. SNC addressed these issues following the staff's requests for additional information.

SNC addressed the issue of SG overfill in the event of an SGTR by examining analyses performed for plants licensed with the methodology developed in WCAP-10698-P-A, "SGTR

Analysis Methodology to Determine the Margin to Steam Generator Overfill," and Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident." SNC concluded that the power uprate and its associated changes should not result in an increase in either the possibility or the probability of SG overfill occurring as a result of an SGTR. The staff finds this consistent with the conclusions drawn in the approved WCAP-10698-P-A methodology on SG overfill.

With respect to flashing fractions in the event of an SGTR, the analysis performed by SNC assumed no flashing fraction for the leakage occurring from the ruptured tube. SNC indicated that some flashing of break flow from the ruptured tube would be expected. However, SNC indicated that they did not consider flashing as part of the FNP licensing basis analysis. SNC did provide figures with examples of break flow flashing fractions as a function of time for facilities that have performed detailed transient analyses and have included flashing as a part of their licensing basis calculation. These transient analyses modeled the expected operator action to terminate primary-to-secondary break flow, including isolation of the faulted SG, cooldown with the intact SGs, and depressurization with the pressurizer PORV.

As provided by SNC, the results in both figures assumed that a loss of offsite power had occurred. One figure assumed that a PORV on an intact SG failed closed while the other figure assumed that the PORV on the faulted SG failed open and required operator action to close the associated PORV block valve. However, SNC did not include in its calculations any assumption for the flashing fraction.

The staff concluded that flashing is real phenomenon and should be included in the assessment of the consequences of the power uprate. Absent detailed flashing information, the staff utilized flashing fractions from a recently reviewed SGTR analysis for an SG replacement. The flashing fractions, which were utilized, would be considered conservative to a degree that the staff believes such values would envelope the flashing that might occur at FNP. With the utilization of these values, doses were found to be acceptable. Nevertheless, as a condition to this amendment, the staff has concluded that SNC should provide an SGTR analysis that incorporates a flashing fraction that is appropriate to the Farley design.

SNC was asked to address the question of tube uncovery during certain transients. SNC indicated that the Westinghouse Owners Group (WOG) had confirmed, on a generic basis, and the staff had concurred in a March 10, 1993, letter that the effects of partial SG tube uncovery has a negligible effect on iodine releases from SGTR and non-SGTR transients. A March 31, 1992, letter from the WOG documented information on the SG tube uncovery issue which had been previously discussed with the staff. This letter stated the following with respect to non-SGTR transients. "For those accidents for which some fuel rods are calculated to be in DNB, the probability of the event and the set of conditions leading to significant offsite releases is considered to be very low (less than 5×10^{-8} per reactor year) for a representative plant. It is possible that plant-specific considerations may modify the transient behavior and change the probability value." In its April 13, 1998, letter, SNC concluded that the assessments done to support the March 31, 1992, WOG conclusion are still applicable considering the effects of the power uprate and the use of ARC. The staff has reviewed SNC's April 13, 1998, letter, and is in agreement with the SNC conclusion.

Table 3.5.1.5-1 presents the assumptions utilized by the staff in its assessment. The potential consequences of an SGTR accident are presented in Tables 3.5.2-1 through 3.5.2-3. The staff concluded that even with the inclusion of the flashing fractions and the assumption that the reactor coolant activity levels were at 60 μCi/g for the preexisting spike and 1 μCi/g for the accident-initiated spike case, doses were found to be acceptable.

3.5.1.6 Locked Rotor

SNC assumed a postulated reactor coolant pump locked rotor event and subsequent leakage of steam from the secondary system due to the leakage of reactor coolant to the secondary system. Leakage from the primary side to the secondary side was assumed to exist at the TS value. SNC assumed that the contribution of activity from the initial reactor coolant activity levels, even when it is assumed that the reactor coolant activity levels are at the TS values for dose equivalent ^{131}I and an iodine spike is considered, is small relative to the contribution of activity from the release of the gap inventory from 20 percent of the fuel rods in the core. SNC indicated that its analysis results showed that fuel melting would not occur during a locked rotor event.

The staff performed independent calculations of the consequences of the locked rotor accident. Table 3.5.1.6-1 presents the assumptions utilized by the staff in its assessment. The staff's assessment of the potential dose consequences of a locked rotor accident are presented in Tables 3.5.2-1 through 3.5.2-3. The doses were found to be acceptable.

It should be noted that the staff performed its assessment of the consequences of this accident based upon primary coolant TS values of dose equivalent ^{131}I of 60 μCi/g and 1 μCi/g. As noted for the SGTR assessment, the calculations were performed at these activity levels based upon the assumption that in the future, upon replacement of the SGs, SNC may wish to restore the TS limits for dose equivalent. Such an assumption will not require the staff to reanalyze this accident in the event that SNC wishes to increase the activity levels of dose equivalent ^{131}I in the reactor coolant at a later date provided that there are no future changes to the various parameters, operator actions, and operator and/or system or reactor responses that would alter the doses calculated for this assessment.

3.5.1.7 Fuel Handling Accident

SNC evaluated two fuel handling accident scenarios. The first assumed the refueling accident occurred within containment. In this scenario, a spent fuel assembly was assumed to be dropped onto the core. This was assumed to result in damage to an entire fuel assembly. Following the drop, the activity released to the containment atmosphere was assumed to be released instantaneously to the environment via the containment purge exhaust filter. SNC assumed that the release via this pathway would have a removal efficiency of 90 percent for the elemental form of iodine and 30 percent for the organic form of iodine. SNC assumed no credit for decay resulting from holdup in the containment.

The second scenario assumed the refueling accident occurred outside containment in the area of the spent fuel pool, which is located in the auxiliary building. As in the previous scenario, it was assumed that the dropping of a spent fuel assembly into the spent fuel pool would result in damage to one entire fuel assembly and the release of the volatile gaseous fission products to the spent fuel pool with subsequent release to the environment through the penetration room filtration system. SNC assumed credit for mixing of activity in the open area just above the spent fuel pool. SNC took no credit for decay due to holdup in the auxiliary building nor credit for decay due to transit time after release to the environment. The penetration room filtration system was assumed to remove iodine with an efficiency of 90 percent when it was in the elemental form and 70 percent when it was in the organic form.

The staff has performed an independent calculation of a fuel handling accident. The staff also evaluated two scenarios. The staff's assessment did not include credit for mixing in the buildings in which the fuel handling accident occurred because the staff concluded that SNC did not provide adequate justification for the assumption of building mixing.

Table 3.5.1.7-1 contains details of the assumptions utilized by the staff in its assessment. The offsite doses are presented in Tables 3.5.2-1 and 3.5.2-2. The control room operator doses are presented in Table 3.5.2-3. The doses were found to meet Part 100 and GDC-19 criteria.

3.5.1.8 Rod Ejection

SNC performed an analysis of a postulated rod ejection accident. It was assumed that the reactor was operating with equilibrium activity levels in the primary and secondary systems based upon 1 percent failed fuel and a primary-to-secondary leak rate of 150 gpd/SG. The analysis was performed with the assumption that there are two potential release pathways following a rod ejection accident. The first pathway is via containment leakage. It is assumed that the rod ejection accident results in the release of activity from the fuel rods to primary coolant. A certain fraction of the activity released to primary coolant is assumed to be released to the containment and available for release from the containment at the TS value for containment leakage.

The second pathway is via the secondary side. For this pathway, the assumption is made that reactor coolant leaks from the primary side to the secondary side and then radioactivity is released to the environment via the relief valves since it is assumed that when offsite power is lost, the accident occurs.

For both release pathways, the assumption is made that all of the gap activity of the fuel that has its cladding degraded as a result of the accident and all of the activity of the fuel that has experienced melting, is released to the primary coolant. SNC's analysis assumed that the rod ejection resulted in 0.25 percent of the fuel in the core melting and 10 percent of the fuel experiencing degradation of cladding.

The staff has performed a calculation of the dose consequences of a rod ejection accident. Table 3.5.1.8-1 presents the assumptions utilized by the staff in its assessment. The doses, which were calculated for this accident, are presented in Tables 3.5.2-1 through 3.5.2-3.

SNC performed the analysis of the consequences of releases occurring from both the secondary side release pathway and the containment pathway. The staff has presented separately the results of the assessment for each path. The thyroid dose at the low population zone exceeded the acceptance criterion of Appendix A to SRP 15.4.8. The staff finds this departure from the acceptance criterion acceptable because:

1. The doses are below the guideline values of 10 CFR Part 100;
2. The consequences were calculated on the basis of fuel failures that are predicted by conservative models; and
3. The SRP acceptance criterion itself is conservative.

All other doses were found to meet the acceptance criterion of SRP 6.4 and SRP 15.4.8.

3.5.1.9 Small Break LOCA

SNC calculated the potential consequences of an SBLOCA. In this assessment, SNP assumed the very same assumptions as those for an LBLOCA with two exceptions. The first exception was the source term that was assumed to consist of 100 percent failure of the fuel cladding that results in release of 100 percent of the gap activity. The second exception assumed that the sprays were not activated. The resultant doses were assumed to occur only from containment leakage with no inclusion of the potential consequences of either a mini-purge or H_2 venting.

The staff performed a similar calculation. The staff's assessment assumed that the particulate form of iodine would undergo removal via plateout. The doses are presented in Tables 3.5.2-1 through 3.5.2-3. The doses were found acceptable.

3.5.2 Conclusions

The staff has assessed those accidents for which the power uprate would have an impact upon the offsite and control room operator doses. These doses are presented in Tables 3.5.2-1 through 3.5.2-3. The staff's results demonstrated that, for those accidents that are impacted by the power uprate, the doses would not exceed the dose guidelines presently contained in the Standard Review Plan, 10 CFR Part 100, or GDC-19 of 10 CFR Part 50, Appendix A, for either offsite locations or control room operators. Therefore, the staff finds the proposed power uprate acceptable.

4.0 SYSTEMS, STRUCTURES, AND COMPONENTS EVALUATION

In its evaluation, the staff refers to several terms in the submittal. These terms are defined below:

- ASME Code - editions of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Sections III or XI, as appropriate, through the 1988 Addenda and the 1989 Edition.

- ASTM Standard Procedures or Practices - procedures or practices for testing or analysis developed by the American Society for Testing and Materials.

- Charpy-V notch test or Charpy-V test - a fracture toughness test that involves impacting a small notched impact specimen (usually in bar form) with an impact pendulum and measuring the fracture energy of the specimen.

- Charpy transition curve or Charpy-V curve - a graphic presentation of Charpy-V test data, including absorbed energy, and fracture appearance. The curve includes the lower shelf energy (< 5 percent shear), the transition region, and the upper shelf energy (> 95 percent shear).

- EOL - "end of life," the scheduled date of expiration of the operating license for a licensed nuclear power generation facility.

- EOL fluence - the best-estimate neutron fluence projected for a specific reactor pressure vessel (RPV) beltline material at the clad-base-metal interface on the inside surface of the RPV at the location where the material receives the highest fluence on the expiration date of the operating license.

- Pressurized thermal shock (PTS) event - an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.

- Reactor pressure vessel (RPV) beltline - the region of the RPV (shell material including welds, heat-affected zone, and plates or forging materials) that directly surrounds the effective height of the active core and adjacent regions of the RPV that are predicted to experience significant neutron irradiation damage (a consideration in selecting materials).

- RT_{PTS} - the reference temperature, RT_{NDT}, evaluated for the EOL fluence for each of the RPV beltline material, using the procedures of Paragraph (c) of the revised PTS rule, 10 CFR 50.61 (Ref. 9). For materials in the RPV beltline, the RT_{PTS} must account for changes in the reference temperature as a result of neutron irradiation damage.

- ΔRT_{NDT} - the transition temperature shift, or change in RT_{NDT}, due to neutron irradiation effects, which is evaluated as the difference in the 30 ft-lb (41 J) index temperatures from the average Charpy-V curves measured before and after irradiation.

- Upper shelf energy (USE) - the average value for all Charpy-V test specimens (normally three) whose test temperature is above the upper end of the transition region of the Charpy-V curve; for specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the USE of the material.

4.1 Reactor Vessel Integrity

To determine the acceptability of the power uprate on the integrity of the reactor vessel, the staff evaluated the following:

- Effect on the EOL USE Values for the Beltline Materials
- Effect on SNC's Revised Pressure/Temperature (P-T) Limit Curves and SNC's Assessment for Prevention Against Pressurized Thermal Shock
- Effect on the Material Surveillance Programs

4.1.1 Effect on the EOL USE Values for the FNP Units 1 and 2 RPV Beltline Materials

Appendix G, "Fracture Toughness Requirements," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50, Appendix G), requires, in part, that the Charpy-V USEs for RPV beltline materials be no less than 75 ft-lb (102 J) in the unirradiated condition, and no less than 50 ft-lb (68 J) throughout the life of the RPV, unless it can be demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of USE (as determined from the results of Charpy-V tests and Charpy-V curves) will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

The staff performed an independent review of the Charpy-V test data for Surveillance Capsule Nos. Y, X, U, and W from Unit 1 and for Surveillance Capsules U, W, and X from Unit 2. These capsules have been removed from the FNP RPVs and tested in accordance with the respective FNP Unit 1 and Unit 2 surveillance withdrawal program requirements set forth in 10 CFR Part 50, Appendix H.

Tables 4.1.1-1 and 4.1.1-2 (Attachment 1) compare the projected EOL USE values determined by the staff and by SNC. SNC's EOL USE values for the FNP Units 1 and 2 beltline materials were revised and submitted to the staff in SNC's submittal of February 12, 1998. Both the staff's and SNC's calculation of the EOL USE values are based on the neutron fluence values for the RPV 1/4T locations as determined from the latest neutron transport calculations for the vessels. These fluences are slightly more conservative than those which would be projected using RPV neutron dosimeter measurements. Tables 4.1.1-1 and 4.1.1-2 show the uprated, limiting EOL USE values for the FNP beltline materials in **bolded print**. SNC's projected EOL USE values for the FNP units under uprated conditions were within -4 ft-lbs to +2 ft-lbs of the USE values calculated by the staff and therefore were in agreement. The tables indicate that the beltline materials in the FNP Units 1 and 2 RPVs will continue to satisfy the EOL USE criteria specified in 10 CFR Part 50, Appendix G, if the request to increase the rated thermal power of the plants is approved by the staff. Conservative neutron-transport-calculation-based fluences were used to project the EOL USE values for the FNP beltline materials.

4.1.2 Effect on SNC's Revised P-T Limit Curves for FNP Unit 1 and Unit 2 and SNC's Assessment for Prevention Against Pressurized Thermal Shock

Section 50.61 to Part 50 of Title 10 of the *Code of Federal Regulations*, requires, in part, that "[f]or each pressurized water reactor for which an operating license has been issued,... the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material...For each pressurized water nuclear reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the

PTS screening criterion using the EOL fluence, the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion...." [1]

In its review, the staff performed its own calculations of the effect of the uprated EOL fluences on the projected EOL RT_{PTS} values for the Units 1 and 2 RPV beltline materials. The staff used the methodology in 10 CFR 50.61 to calculate projected RT_{PTS} values. The review is documented in the staff's SE dated March 31, 1998 (Reference 21).

In that SE, the staff determined that the RT_{PTS} values will continue to satisfy the criteria of 10 CFR 50.61 under the uprated conditions for the plants. Therefore, the staff concludes that after the power uprate, FNP Units 1 and 2 will remain in compliance with the criteria of the revised PTS rule, 10 CFR 50.61.

The staff also assessed SNC's requests for approval of the uprated P-T limit curves (PTLRs) for the RCSs, and of SNC's proposed PTLRs for the FNP facilities. Appendix G to 10 CFR Part 50 states the requirements for generating P-T limit curves; Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," provides the staff's position on removing the P-T limit curves from the limiting conditions for operation in the TS. In its SE (Reference 21), the staff determined the proposed, uprated P-T limit curves and PTLRs for the FNP RPVs were acceptable for implementation. Therefore, the staff concludes that the power uprate will not affect SNC's compliance with the criteria of Appendix G to 10 CFR Part 50 or conformance with the staff's position stated in GL 96-03.

4.1.3 Effect on the Material Surveillance Programs for the FNP Units 1 and 2

Appendix H to 10 CFR Part 50 provides the NRC's requirements regarding licensee-implemented RPV material surveillance programs. SNC has provided the surveillance capsule withdrawal schedules for the FNP units in Table 3-1 of the respective PTLRs for the units. In its submittal of July 13, 1997, as amended by its submittal of December 18, 1997, SNC requested a number of changes to the FNP Units 1 and 2 material surveillance programs among its PTLR license amendment requests. The proposed changes are based on the uprated neutron fluence values for the beltline materials and surveillance capsules. SNC has informed the NRC that it is currently using the criteria of ASTM Standard Practice E185-82 as the current basis for its material surveillance programs for the FNP units. [2] This is consistent with the criteria of 10 CFR Part 50, Appendix H. The staff's evaluation of the proposed changes to the surveillance programs is provided in an SE (Reference 21). In this SE, the staff reviewed the proposed changes to the materials surveillance programs as proposed in Table 3-1 of the proposed PTLRs for the units. The staff determined that the changes to limiting neutron fluence levels for the surveillance capsules under the power uprate will not affect the ability of the materials surveillance programs to comply with the requirements of 10 CFR Part 50,

---

1 According to the revised rule, 10 CFR 50.61, the PTS screening criteria are 270°F for plate materials, forging materials, and axial weld materials, and 300°F for circumferential weld materials.

2 Henceforth ASTM Standard Practice E185-82 will be abbreviated as E185-82.

Appendix H. Therefore, the staff concludes that SNC's proposed changes to the material surveillance programs for FNP Units 1 and 2 are acceptable.

4.2 Reactor Vessel

SNC reported that the power increase will result in changing the design parameters given in Table 2.1-1 of Reference 36. Table 2.1-2 provides various cases that were developed for use in the power uprate analysis.

SNC evaluated the reactor vessel considering the worst load sets of operating parameters and design transients and a new set of LOCA loads, which was identified for the uprated power condition. The regions of the reactor vessel affected by the power uprate include outlet and inlet nozzles, the RPV (main closure head flange, studs, and vessel shell), control rod drive mechanisms (CRDM) housing, bottom head to shell juncture, core support pads and the instrumentation tubes. SNC evaluated the maximum ranges of stresses and cumulative fatigue usage factors for the critical components at the core power uprated conditions. The evaluation was performed in accordance with the ASME Code, Section III, 1968 Edition, with addenda through the Summer 1970 to assure compliance with the code of record.

The calculated maximum stresses and the maximum cumulative fatigue usage factors (CUF) for the reactor vessel critical locations are provided in Table A of Reference 6. The results indicate that the maximum stresses are within the allowable limits for FNP Units 1 and 2, and the CUFs remain below the allowable ASME Code limit of 1.0.

On the basis of its review, the staff concludes that the current design of the reactor vessel is in compliance with codes and standards under which the plant was licensed, for the uprated power conditions.

4.3 Reactor Core Support Structures and Vessel Internals

By letter dated August 5, 1997, SNC provided additional information requested by the staff, with regard to the evaluation of the reactor vessel core support and internal structures. The limiting reactor internal components evaluated include the lower core plate, core barrel, baffle plate, baffle/barrel region bolts, and the upper core plate.

SNC evaluated the upper and lower internals considering the worst case set of operating parameters provided in Table 2.1-2 of Reference 36. The evaluation was performed in accordance with the original design-basis criteria for the FNP reactor internals, which had been previously reviewed by the staff. For the lower core plate reanalysis, the evaluation was performed using Section III, Subsection NG of the ASME Code, 1989 Edition with 1990 Addenda. The maximum calculated stresses and cumulative fatigue factors for the limiting internal components at the power uprate conditions are identified on page 45 of Reference 4. The maximum stresses for components were below the Code allowable limits, except the upper core plate and the baffle/barrel bolts. For the upper core plate, the maximum combined primary and secondary stress exceeded the 3Sm limit. A simplified elastic-plastic analysis was performed in accordance with ASME NG-3228.3. The reanalysis indicated that the fatigue usage factor was 0.08, which is less than the limit of 1.0, and that the combined primary and secondary stress intensity, excluding thermal bending stresses, was less than 3Sm. No stress was given for the baffle/barrel bolts. SNC indicated that the baffle/barrel bolts were originally qualified by test to loads associated with the existing design basis conditions, and these bolts are acceptable because the existing design basis condition is bounding for the proposed power update condition. The maximum CUF was calculated to be 0.917 for the baffle/barrel bolts at the proposed power uprate condition.

It is noted that baffle/barrel bolts degradation was reported in European PWR plants. The Westinghouse Owners Group (WOG) has identified a lead plant for inspection in the fall of 1998 for a possible baffle/barrel bolt cracking. SNC should assess the results of the lead plant's inspection for its applicability to the baffle/barrel bolts at FNP and update its evaluation and conclusion as necessary.

Further, SNC reviewed the potential for a flow-induced vibration on the guide tubes and the upper support column at the uprated power level. The evaluation indicated that the existing analysis provides sufficient margins to accommodate the increase (by approximately 1.9 percent) in the flow-induced vibration loads due to the power uprate.

On the basis of the above evaluation, the staff concludes that the reactor internal components at FNP Units 1 and 2 will retain within the allowable limits of stress and fatigue usage factor for operation at the proposed uprated power conditions.

4.4 Control Rod Drive Mechanisms

SNC evaluated the adequacy of the CRDMs by reviewing the FNP current Model L106A CRDM design specifications and stress report to compare the design-basis input parameters against the operating conditions at the uprated core power. The components reviewed include the full length (F/L) L-106 CRDMs, the Capped Latch Housing Assembly and the Royal Industries Part Length (P/L) CRDMs employed at FNP. The maximum temperature for FNP power uprate is 613°F, which is below the design temperature of 650°F. The pressure remains the same at 2250 psia. On the basis of this evaluation, SNC concluded that the original design basis thermal and structural analyses are bounding for the core power uprate.

On the basis of its review, the staff concludes that the current design of CRDMs is in compliance with licensing basis codes and standards for the uprated power conditions.

4.5 Steam Generators

SNC evaluated the SGs by comparing the power uprate conditions with the design parameters of the Westinghouse Model 51 SGs at FNP. The comparison shown in Table 2.1-2 of Reference 36 indicates that critical design system parameters such as the primary and secondary side pressures, as well as the vessel outlet and secondary side temperatures, are not significantly affected by the uprated power conditions. The variation in the primary-to-secondary pressure differential is within about 8 percent. SNC indicated that there are no significant changes to the design transients as a result of the core power uprate. The evaluation was performed in accordance with the requirements of the ASME Code, Section III, 1971 Edition.

The maximum stresses and cumulative fatigue usage factors for the critical SG components are provided in Table B, page 53 of Reference 6. The results indicate that the maximum calculated stresses are below the Code allowable limits except for the divider plate. For the divider plate, SNC performed a plastic analysis considering the actual material stress-strain relation and the stress redistribution, in accordance with the NB-3228.1(b) of the 1971 ASME Code. The analysis indicated that the maximum strain and the calculated CUF are within the allowable range, and that the divider plate will remain in compliance with licensing basis codes and standards for the uprated power condition. The revised CUF values for other SG components excluding the U-bend tubes are less than the Code limit of unity except for the secondary manway bolts at which the CUF was calculated to be 1.18. SNC indicated that it plans to replace the secondary manway bolts prior to 34 years of service so as to remain within compliance with the fatigue limit requirement.

SNC assessed the potential for flow-induced vibration for the small radius U-bend tubes at the power uprate condition. The evaluation concluded that changes to vibration levels and cross flow velocity due to power uprate were within the acceptable range. The U-bend fatigue evaluation was updated using the FNP existing methodology in WCAP-11875 (Reference 32). The results indicate that the fatigue usage of the U-bend will exceed the acceptance limit of 1.0 in 13.7 years after the implementation of the power uprate at FNP Units 1 and 2 in 1998 outages. SNC concluded that the U-bend needs to be monitored and would require some type of corrective action at that time, as necessary. SNC noted in Section 5.7.3 of Reference 36 that following the implementation of power uprate, the outlet pressure and the steam flow for each steam generator can be documented on a cycle-specific basis for use in any future update of the U-bend fatigue evaluation.

On the basis of its review, the staff finds that the current FNP Units 1 and 2 SGs are acceptable for the proposed core power uprate up to 13.7 years following its implementation at both Units 1 and 2. It should be noted that SNC currently is planning to replace the Unit 1 SGs during the spring 2000 refueling outage and Unit 2 SGs during the spring 2001 refueling outage. Upon replacement, the issue regarding exceeding the cumulative fatigue usage factor for the secondary manway bolts and U-bend tubes will no longer exist.

4.5.1 <u>SG Tube Integrity</u>

4.5.1.1 <u>SG Tube Degradation</u>

To minimize the effect of the power uprate on tube degradation, SNC intends to maintain the same T_{hot} (the hot leg temperature) before and after power uprate, with a 0.5 °F allowance to account for measurement variations and uncertainties. Industry experience has shown that, in general, a high T_{hot} correlates with increased tube degradation. Therefore, limiting T_{hot} to the preuprate value should prevent the rate of tube degradation from increasing after the power uprate. However, the power uprate will slightly reduce the steam pressure and the corresponding saturation temperature, changes that may increase tube degradation.

SNC evaluated the effect of power uprate on tube degradation mechanisms, including primary water stress corrosion cracking (PWSCC), and outside diameter stress corrosion cracking

(ODSCC), in various regions of tubing. The staff's assessment of the mechanisms is discussed below.

4.5.1.1.1 PWSCC of tubing

SNC stated that the main contributors to PWSCC are residual, pressure, and thermal stresses. SNC stated that the power uprate would increase only the through-wall pressure stress. This stress would increase because the primary-to-secondary differential pressure would increase from about 1435 psi to 1463 psi. The resulting pressure stress will increase the PWSCC kinetics by 2 percent to 2.5 percent. SNC concluded that considering the uncertainties in estimating the various contributions to PWSCC, the increase in PWSCC kinetics is insignificant; therefore, the effect of the power uprate on PWSCC is insignificant. The staff finds that an increase of 2 percent to 2.5 percent in PWSCC kinetics is insignificant and, therefore, the impact of the power uprate on PWSCC of tubing is insignificant.

4.5.1.1.2 ODSCC of tubing

SNC evaluated the regions where ODSCC has occurred, such as in the tube support plate crevices, the sludge pile at the top of the tubesheet, and the free span regions of the tubing. SNC found that the beneficial effect of lowering secondary temperature tends to be stronger than the deleterious effect of increasing the applied stress in the tube support plate crevices and on the free span regions. Therefore, the power uprate will minimally affect the tube support plate crevices and free span regions. However, SNC predicted the expected ODSCC rate in the sludge pile to slightly increase because its analysis did not take credit for the lower secondary temperatures in the sludge pile. SNC inspects the parts of the tubes in the sludge pile during each refueling outage and would readily detect any increase in tube degradation there. Based on SNC's assessment, the staff finds that the effect of power uprate on ODSCC of tubing would be minimal.

4.5.1.1.3 Degradation of U-bends in small radius tubes by stress corrosion cracking

SNC stated that it stress-relieved the small radius U-bends to reduce their residual stresses. The increased wall pressure stress and the reduced residual stress cancel each other out so that the impact of the power uprate in the small radius U-bends is negligible. To ensure stress corrosion cracking does not increase in the small radius U-bends, SNC inspects all U-bends in rows 1 and 2 at the refueling outage after the power uprate. Based on SNC's assessment, the staff finds that the impact of the power uprate on stress corrosion cracking in the small radius U-bends is insignificant.

4.5.1.1.4 Tubes susceptible to wear from anti-vibration bars

SNC stated that it replaced the anti-vibration bars on steam generator tubing in Units 1 and 2. Since replacing them, SNC has not observed any tube wear typically caused by anti-vibration bars. SNC stated that the power uprate is not expected to cause wear from the replaced anti-vibration bars. Based on SNC's observation, the staff finds that the power uprate should not significantly affect the tube wear by anti-vibration bars.

4.5.1.1.5 Additional surveillance to monitor tube degradation

SNC stated that before each SG inspection, it assesses degradation mechanisms active in the FNP SGs and in similar SGs throughout the industry. SNC plans to develop inspection plans that will ensure adequate detection of the degradation mechanism in the affected areas. Any new degradation mechanisms and significant increase in degradation rates should be detectable by the planned inspections. The staff is satisfied with SNC's plan to assess the inspection plan for the FNP SGs should new degradation mechanisms be discovered.

4.5.1.1.6 Minimum tube wall thickness

SNC determined the minimum required wall thickness to be 0.022 inch (44 percent of wall thickness) on the basis of a safety margin of three against failure by bursting during normal operations (Reference 27). SNC stated that the plugging limit of 40 percent through wall in the current FNP TS satisfies the required minimum wall thickness under the power uprate conditions. SNC also stated that the 40 percent through-wall plugging limit applies only to tube degradation detected by a qualified sizing technique. The staff finds that the minimum tube wall thickness will maintain the structural integrity of the tube under power uprate conditions.

Based on its review of the information submitted, the staff concludes that the structural integrity of steam generator tubing is acceptable under the power uprate conditions.

4.5.2 SG Tube Plugging and Repair Criteria

SNC evaluated the effect of the power uprate on the ODSCC of the tubes in the tube support plate intersections under the voltage-based alternate repair criteria in the FNP TS. SNC implemented the alternate repair criteria in accordance with the guidelines of GL 95-05 (Reference 26). The criteria were approved by the staff on March 24, 1997, for Unit 1 (Reference 42) and October 11, 1996, for Unit 2 (Reference 41). SNC determined that the voltage-based repair criteria are not affected by the increase in the primary-to-secondary differential pressure associated with the power uprate under normal conditions. Under the alternate repair criteria, tubing is qualified for a safety margin of 1.4 against steamline break differential pressure, a margin that is not changed by the power uprate. SNC also stated that the upper voltage repair limit of the alternate repair criteria is not directly affected by the power uprate. Based on SNC's assessment, the staff finds that the power uprate has an insignificant impact on ODSCC of the tubing under voltage-based alternate repair criteria.

SNC stated that the only repair criteria that will be affected by the power uprate are the F* criteria for Unit 2. Unit 1 has no F* criteria. The F* criteria in Unit 2 TS 3/4.4.6 are affected by the increased differential pressure between the primary and secondary loops under the power uprate. SNC proposed revising the existing F* distance from 1.54 inches to 1.6 inches in the FNP Unit 2 TS. The F* distance is the length of the expanded part of the tube inside the tubesheet that provides a sufficient length of undegraded tube expansion to resist the pullout of the tube from the tubesheet. Under the power uprate, the F* distance needs to be lengthened to provide additional tube friction surfaces to resist the increase of the pullout force resulting from increased primary-to-secondary differential pressure during postulated accidents. Based on the staff's previous review, the staff determined that the proposed F* distance of 1.6 inches is acceptable and may be incorporated into the FNP Unit 2 TS.

4.5.3 SG Blowdown (SGBD) System

The SGBD system is used to control the chemical composition and buildup of solids in the SG shell water. At FNP, the SGBD can handle a continuous blowdown rate of 12.5 gpm per SG and a maximum intermittent blowdown rate of 50 gpm per SG. The actual blowdown rates during plant operation are based on chemistry control and the requirements for controlling solid buildups on SG tubesheets. These parameters will not be significantly affected by power uprate and the blowdown rates for power uprate conditions will be bounded by the current rates. The staff finds that the existing SGBD rates will remain unchanged for the conditions resulting from the power uprate.

4.6 Reactor Coolant Pumps (RCPs)

SNC evaluated the FNP RCPs in accordance with Section III, Subsection NB of the ASME Code, 1971 Edition, with Addenda through Summer 1972. The evaluation was performed by reviewing the design analysis of the Westinghouse Model 93A RCPs in comparison with the proposed uprated conditions as shown in Table 2.1-2 of Reference 36. The components evaluated included the RCP casing, main flange bolts, thermal barrier, casing foot, discharge and suction nozzles, casing weir plate, seal housing, and auxiliary nozzles.

After the core power uprate, the RCS pressure remains unchanged. There are no significant changes to the design thermal transients. The number of occurrences of applicable transients in 40 years are provided in a table on page 50 of Reference 6. The increase in temperature fluctuation due to power uprate for the RCP was used in the stress analysis for components that constitute the pressure retaining boundary. The evaluation indicated that the maximum stress intensity in the casing weir plate was less than the allowable Code limit. The calculated CUFs were less than the fatigue limit of 1.0.

On the basis of its review, the staff finds that the current Model 93A RCPs, when operating at the proposed power uprated conditions, will remain in compliance with the requirements of the codes and standards under which the FNP Units 1 and 2 were originally licensed.

4.7 Pressurizer

SNC evaluated the adequacy of the pressurizer and components including the pressurizer spray nozzle, safety and relief nozzle, upper head and shell, manway pads and bolts, instrument nozzle, support lug, surge nozzle, lower head/heater well, immersion heater, and valve support bracket for operation at the uprated conditions. The evaluation was performed in accordance with the requirements of the ASME Code, Section III, 1968 Edition, through Summer 1970 Addenda, for FNP Unit 1 and 1968 Edition, through Winter 1970 Addenda for FNP Unit 2.

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg (T_{hot}) and cold leg (T_{cold}) temperatures are low. This maximizes temperature differential and thermal stress that are experienced by the pressurizer. The RCS pressure is unchanged for the FNP power uprate. There is an insignificant change in the design transients regarding type and number of occurrences during the 40 years of plant operation, for FNP power uprate as seen in a comparison table on page 57 of Reference 6. The minimum cold leg temperature decreased by 13°F, with respect to the original design conditions. The original fatigue analyses were updated to account for the uprated power conditions. The CUFs at the critical locations are provided in a table on page 58 of Reference 6. SNC indicated that the maximum calculated stresses at the critical locations are unchanged except for the surge line nozzle. The maximum stress intensity at the surge line nozzle was recalculated and was found to be less than the allowable Code limit. The maximum CUFs at the limiting locations are 0.94 for the surge nozzle and 0.78 for the pressurizer upper head and shell.

The performance of the pressurizer safety valves (PSVs) and the PORVs are dependent on the pressurizer operating pressure and temperature, which are unchanged for the power uprate at FNP. The transient analysis for the power uprate was performed assuming a maximum tolerance of lift setting, as specified in Sections 3/4.4.2 and 3/4.4.3 of the FNP Technical Specifications. The PSVs and PORVs setpoints, rated capacities, and corresponding dynamic loads imposed on the piping and supports due to valve operation do not change as a result of the proposed power uprate.

On the basis of the above review, the staff finds that the existing pressurizer and components remain adequate for the plant operation at the proposed uprated core power.

4.8 Chemical and Volume Control System (CVCS)

The main role of the CVCS is to maintain RCS water inventory, boron concentration, and water chemistry. To perform these functions, the CVCS must meet the following requirements: (1) the parts of the system that constitute the reactor coolant pressure boundary can withstand the expected RCS conditions, (2) boration meets the design requirements for reactivity control, and (3) with the exception of RCP seal injection line, the system can be automatically isolated during all events requiring its isolation. The proposed power uprate will not affect the CVCS isolation function, but it will have some effect on the integrity of the RCS pressure boundary and on the boration of the RCS. SNC therefore, evaluated the effect of the uprate on the performance of the CVCS.

4.8.1 RCS Temperature

During power uprating, a change in reactor coolant temperature may affect the integrity of the primary coolant boundary during normal operation and during thermal transients. However, SNC's evaluation concluded that, after the power uprate, the design basis RCS cold leg temperature of 541.1°F will be well below the system design temperature of 650°F and below the maximum RCS inlet operating temperature of 547°F, established in the CVCS overall design. Furthermore, the regenerative, letdown, excess letdown, and seal water heat exchangers will also operate within acceptable limits, and the load on the component cooling water system will not be excessive. The CVCS will, therefore, preserve its design functions. SNC also analyzed the performance of the CVCS during conditions resulting from thermal transients and found that it is bounded by the original design parameters. The staff finds SNC's evaluation acceptable.

4.8.2 Boration

A revised average RCS temperature range after the power uprate will result in a decrease of the RCS volume, and a smaller volume will make the boration operation more conservative. However, boron dilution will occur at a faster rate and would adversely affect the RCS shutdown margin. SNC analyzed an uncontrolled boron dilution event for the uprated power level to identify the minimum amount of time that would be available to terminate an inadvertent boron dilution before a complete loss of shutdown margin. The analysis showed that for both automatic and manual control modes; the boron dilution could be terminated in time to avoid exceeding the safety margin. The staff finds SNC's evaluation acceptable.

The staff finds the operation of the CVCS at uprated power conditions acceptable because all the requirements for its satisfactory performance are met.

4.9 NSSS Piping and Pipe Supports

The proposed power uprate of FNP Units 1 and 2 involves the increase of temperature difference across the RCS. SNC evaluated the NSSS piping and supports by reviewing the design basis analysis against the uprated power condition, with regards to the design system parameters, transients, and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop (RCL) piping, primary equipment nozzles, primary equipment supports, pressurizer safety and relief valve (PSARV) piping, and the pressurizer surge line piping. The methods, criteria, and requirements used in the existing design basis analysis for FNP Units 1 and 2 were used for the power uprate evaluation.

The RCS pressure remains unchanged for the proposed core power uprate. The actual hot leg temperature for the power uprate is projected to be equal to or less than the hot leg temperature at the current rated power level. The cold leg temperature for the power uprate condition will be less than for the current power level. The changes in design transients are insignificant for FNP power uprate with regards to the type and number of occurrences during the 40-year plant operation. The loop hydraulic forces will increase slightly due to the decrease in the cold leg temperature and the increase in water density at the power uprate condition.

The original structural design basis of FNP RCL piping and supports required consideration of dynamic effects resulting from postulated guillotine breaks in the primary loop piping. SNC's leak-before-break (LBB) analyses for the FNP Unit 1 and Unit 2 reactor coolant systems is provided in Westinghouse WCAP-12825 (Reference 33). In this analysis, Westinghouse indicated that the ratio of the critical flaw size to the leakage flaw size was 4.1 for the FNP Unit 1 RCS piping. The corresponding ratio for the FNP Unit 2 RCS piping was 5.6. Since both ratios exceeded the required safety margin of 2.0, the staff approved SNC's RCS piping LBB analysis for implementation in its letter of August 12, 1991 (Reference 37).

The staff determined that the proposed 4.6 percent increase in reactor power for the FNP units would only increase the pressure and thermal loadings for the RCS piping. The proposed power uprate will not affect the dead weight, safe shutdown earthquake (SSE), and seismic anchor motion (SAM) loadings. The staff conservatively estimates that the resulting increase in the total loads for the LBB analysis would not reduce the ratio of the critical flaw size to leakage flaw size by more than 10 percent of the values reported in WCAP-12825. These margins are still considerably greater than the minimum requirement for a safety margin of 2.0 on RCS piping LBB applications. Therefore, the proposed power uprate will not change any of the LBB conclusions in stated in WCAP-12825, or in the staff's safety evaluation of August 12, 1991. In addition, postulated breaks in the primary loop piping have been eliminated and replaced with postulated breaks in the surge line on the hot leg and in the accumulator line on the cold leg. As a result of the elimination of primary loop piping breaks, the existing margin (approximately 30 percent), based on consideration of the accumulator line and surge line breaks, is sufficient to accommodate the increase in loop hydraulic force due to the power uprate.

SNC's evaluation indicated that in all cases, except for the RCL crossover leg usage factor, the existing design basis stress analyses for the RCS system piping and supports and systems connecting to the RCS system, are bounding for the power uprated conditions. The crossover leg CUF experiences a minor increase but remains below the specified limit for a postulated break. The evaluation was performed in accordance with Section III of the ASME Code, to assure compliance with the code of record at FNP Units 1 and 2. SNC concluded that the plant power uprate has no adverse effect on the ability of these components to operate.

The staff finds that the increase in temperature difference across the RCS system, will not have adverse effect on the NSSS piping, and will minimally impact the design basis analysis of the piping and pipe supports. Therefore, the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the branch lines connecting to the primary loop piping will remain in compliance with the requirements of the design bases criteria, as defined in the FSAR, and are acceptable for the power uprate.

4.10 NSSS/BOP Interface Systems

4.10.1 Auxiliary Feedwater System/Condensate Storage Tank

SNC performed evaluations of the effects of plant operations at the proposed uprated power level on auxiliary feedwater (AFW) system/condensate storage tank (CST). It was determined that the AFW system components have sufficient margin to provide the required flow and pressure. The condensate inventory required for the AFW system CST volume was determined to be 131,000 gallons for plant operations at the proposed power level. The current CST has a 500,000 gallon capacity. The condensate makeup suction is located at an elevation that ensures a reserve of 150,000 gallons for emergency decay heat removal with the AFW system. The AFW pump intake lines are located close to the bottom of the CST and provide sufficient useful CST volume to bound the above cited design requirement.

Based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on the AFW system/CST.

4.10.2 Component Cooling Water System (CCWS)

The CCWS is a closed loop system that serves as an intermediate barrier between the service water system and systems that contain radioactive or potentially radioactive fluids in order to eliminate the possibility of an uncontrolled release of radioactivity. It provides cooling water to various safety and nonsafety systems during all phases of normal plant operation, including startup through cold shutdown and refueling, as well as following an SBO event, LOCA, or MSLB accident. The CCWS heat loads resulting from plant operations at the proposed uprated power level will increase slightly. SNC performed evaluations of the effects of these increases in heat loads on CCWS and stated that the additional heat loads result in minor temperature increases on the CCWS for normal and accident scenarios and that the CCWS has the capacity to accommodate the additional heat loads and resultant temperature increases. No increase in CCWS flow rates is required to handle the additional heat loads.

Based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants, the staff finds that plant operations at the proposed uprated power level do not change the design aspects and operations of the CCWS and have an insignificant or no impact on the CCWS. Therefore, the staff concludes that the FNP CCWS is acceptable for operations at the proposed uprated power level.

4.10.3 Spent Fuel Pool Cooling System (SFPCS)

The SFPCS is designed to remove the decay heat released from the spent fuel assemblies stored in the spent fuel pool (SFP); to maintain the SFP water temperature at or below the maximum operating temperature limit of 150°F during plant operations and refueling; and to maintain its cooling function during and after a seismic event. The SFPCS heat loads will increase slightly resulting from plant operations at the proposed power level.

SNC performed SFP heatup analyses for plant operations at the proposed power level. For the bounding (emergency full core off-load)[3] case, the SFPCS heat load increases from 34.77 x 10^6 Btu/hr to 37.0 x 10^6 Btu/hr, and the corresponding calculated peak SFP temperature with one SFP cooling train in service increases from less than 170°F to 175°F. With two[4] SFP cooling trains in operation the SFP can be maintained below 140°F.

SNC stated that FNP operates the SFP cooling system with one SFP cooling train in service and the remaining train on standby. Plant administrative procedures, "Outage Nuclear Safety" and "Outage Planning Manual" have provisions to ensure that the backup SFP cooling train is available prior to the outage. Plant procedures also require fuel handling operations be suspended upon receipt of the high SFP temperature alarm which is set at 130°F. Plant procedures provide the controls necessary to ensure that the maximum operating temperature limit of 150°F will not be exceeded. Nevertheless, SNC performed evaluations (with demineralizer isolated and fuel handling operations prohibited) to demonstrate the acceptability

---

3 The maximum SFPCS heat load for the end-of-cycle full core off-load during routine refueling is 36.5 x 10^6 Btu/hr. The corresponding calculated peak SFP temperature is 174°F with one SFP cooling train in service.

4 Simultaneous operation of both SFPCS trains is not a normal practice at FNP.

of SFPCS operation, SFP liner and concrete structure at temperatures below 180°F. SNC concluded that no changes to the SFP cooling or cleaning systems are required to support plant operations at the proposed power uprate level.

In the unlikely event that there is a complete loss of SFPCS cooling capability, the SFP water temperature will rise and eventually will reach boiling temperature. SNC performed analysis to demonstrate the time to boil and the boil off rate based on the heat load for the emergency full core off-load scenario. The calculated minimum time from the pool high temperature (130°F) alarm caused by loss-of-pool cooling until the pool boils is 5 hours and the maximum boil-off rate is 76.4 gpm. However, makeup water from the primary water storage tank can be initiated within the required time utilizing either of two 165 gpm reactor water makeup pumps. The primary water storage tank, reactor water makeup pumps, and piping are seismic Category I. In addition, makeup water can be supplied from the refueling water storage tank and the demineralized water system.

In addition, the amount of fission product release and the chemical and radionuclear composition of pool water will not change appreciably after the power uprate and the existing cleanup system will perform adequately. The cleanup system will be able to withstand the temperatures of pool water for partial and full core off-loads, as long as the two cleanup trains are in operation. When only one train is in operation, the temperature of bulk water will exceed 140°F, the upper limit for the demineralizer resin. However, the system has a provision to alarm the operators whenever water reaches 130°F, so that they can manually initiate appropriate actions to prevent damage to the cleanup system. The staff finds that with this provision, the existing SFP cleanup system will be adequate for maintaining purity of the SFP water after the power uprate.

Based on its review, the staff finds the SFPCS acceptable for plant operations at the proposed power uprate level.

4.11 Main Turbine

SNC performed evaluations on turbine operations with respect to design acceptance criteria to verify the mechanical integrity under the conditions imposed by plant operations at the proposed uprated power level. Results of the evaluations showed that there would be no increase in the probability of turbine overspeed. Therefore, the turbine could continue to be operated safely at the proposed uprated power levels.

Based the staff's review and the experience gained from its review of power uprate applications for similar PWR plants, the staff finds that operation of the turbine at the proposed uprated power level is acceptable.

4.12 High Energy Line Break (HELB) Outside Containment

System operating parameters for uprate were evaluated against the system pressure and/or temperature parameters used in the existing plant bases to demonstrate the acceptability for HELB effects. Core uprate will not change the bounding temperature and pressure used as the basis for pipe break analyses. SNC stated that design basis analyses remain bounding for all HELB events.

Based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level have an insignificant or no impact on the consequences (e.g., environmental pressure and/or temperature parameters, etc.) resulting from HELB outside containment.

4.13 Safety-Related Equipment Qualification (EQ)

The effects of all changes due to plant operations at the proposed uprated power level on design and EQ of mechanical components were evaluated. The temperatures, pressures, and in some cases flows, in certain systems would be affected slightly by plant operations at the proposed uprated power level. However, these changes in temperatures, pressures, and flows are bounded by the original design of components. The existing parameters used for qualifying mechanical components inside and outside containment remain bounding for the conditions resulting from plant operations at the proposed power level.

As part of the safety-related electrical equipment qualification evaluation for the power uprate, SNC reviewed: (1) the radiological dose limit for safety-related electrical equipment located in a harsh environment (e.g., containment), (2) the composite temperature and pressure curves for safety-related electrical equipment qualification, and (3) components that were qualified based on calculated surface temperatures.

(a) Radiological doses

SNC's review of the radiological doses for the power uprate showed that many of the original design-basis doses were bounding. For safety-related electrical equipment not bounded by the original design basis, radiological doses at uprate conditions were compared with the dose threshold limits used for the individual components or individual items of equipment. SNC finds that the comparison showed sufficient margin available to accommodate the increased uprate dose without compromising the equipment qualification.

The staff raised a question that the dose should be bounded by the test report values, not by the dose threshold limits. By letter dated August 5, 1997, SNC clarified that the terminology of dose threshold limits and test report values have the same meaning in the context of equipment qualification evaluation prepared for power uprate.

SNC concludes that the radiological cumulative dose or dose rate was either enveloped by the results of previous design-basis radiological analysis or was within the threshold limit for which the individual components or individual items of equipment were qualified.

(b) Containment pressure and temperature evaluation

SNC's comparison of the composite uprate temperature profile to the existing composite temperature profile indicated that the uprate maximum accident temperature is approximately

10°F less than the existing maximum design-basis accident temperature and that the composite uprate temperature profile is enveloped by the existing design-basis composite temperature profile except during the first 70 seconds and after 30,000 seconds.

During the first 70 seconds, the uprate MSLB temperatures are higher than the existing MSLB temperature. Since the higher MSLB temperatures last only a relatively short time, and considering the thermal lag time associated with increasing the temperature of the containment, SNC states that the initial higher temperature would have no significant effect on the qualification of safety-related electrical equipment.

For more than 30,000 seconds, the uprate temperatures exceed the existing design basis profile by a few degrees (approximately 5°F). SNC states that the safety-related electrical equipment qualification test temperatures have enough margin to envelop the higher temperatures because there is adequate margin between the uprate composite profile and the existing design basis composite profile between 70 and 10,000 seconds to compensate for the time when the uprate composite is slightly higher than the existing design-basis profile.

In response to the staff's question about whether composite temperature profiles of the proposed power uprate should be extended further (i.e., beyond 30,000 seconds), by letter dated August 5, 1997, SNC furnished the composite temperature profiles (before and after power uprate) that extend to 30 days and provided more details by dividing the time line into three sections: namely, the initial 150 seconds, 150 seconds to 7,000 seconds, and 7,000 seconds to 30 days. For the initial 150 seconds, based on engineering experience with transient thermal heat transfer analysis (thermal lag analysis), the ramp-up due to exposing the equipment to a high temperature for a short duration would not heat up instantaneously and would not reach thermal equilibrium due to thermal transfer from the environment to the equipment surface. Thus, SNC determined that the short duration is covered by the existing test data. As for the intermediate section, the power uprate composite temperature profile is enveloped by the existing composite temperature profile. For the last section that the power uprate composite temperature profile exceeds the existing composite temperature profile by approximately 5°F, SNC stated that a review of the test profiles for equipment inside the containment indicates that sufficient margin exists in the test profiles to envelop the power uprate composite temperature profile. By letter dated September 22, 1997, SNC furnished two test profiles (i.e., typical of a short-duration and typical of a 30-day test), which showed an adequate margin between the test profiles and the power uprate composite temperature profile.

In addition, SNC finds that the composite uprate pressure profile is enveloped by the existing design-basis composite pressure profile. Thus, SNC finds that the impact of temperature and pressure conditions as a result of MSLBs or a LOCA at power uprate conditions remains bounded.

On this basis, SNC concludes that the environmental qualification testing of safety-related electrical equipment that envelops the existing composite temperature profile adequately envelops the composite uprate temperature profile. However, for typical short-duration test, the staff noted that an equivalency evaluation using Arrhenius methodology is performed (in accordance with the guidelines provided by EPRI and Regulatory Guide 1.89, Rev. 1, Section C.5.b) to extend the test duration to the FNP-specific accident requirements. As a separate initiative outside the scope of this evaluation, the staff will continue to review the adequacy of an equivalency evaluation using Arrhenius methodology.

(c) Surface temperature analyses

Comparison of the MSLB cases used in the current design-basis surface temperature analyses with the corresponding uprate MSLB cases indicated that the surface temperature results are bounding and are not affected by the power uprate. SNC also compared the design-basis post-LOCA containment temperature profile with the uprate post-LOCA containment temperature profile and finds that the original temperature curve was bounding and that the power uprate does not affect the surface temperature.

SNC finds that maximum temperature for the MSLB cases used in the design-basis surface temperature analyses are greater than or comparable with the corresponding uprate MSLB cases.

Based on its review, the staff concludes that plant operation at the proposed uprated power level will have an insignificant or no impact on the EQ of safety-related mechanical components inside or outside the containment. In addition, the existing safety-related electrical equipment qualification is not affected by the power uprate and remains bounded for the power uprate, and therefore, is acceptable.

4.14 Safety/Relief Valves

SNC indicated that relief valve setpoints have been assumed to be 103 percent of the lift setting in the analysis for the power uprate. The relief valve setpoints, rated capacities, and corresponding dynamic loads due to valve operation imposed on the piping and adjacent structures do not change as a result of uprate. On the basis of its review, the staff finds the safety and relief valves will continue to perform their function at the power uprate condition.

4.15 Reactor Trip System/Engineered Safety Feature Actuation System Instrumentation Trip Setpoints and Allowable Values

SNC proposed one change to Trip Setpoint in TS Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, and two changes to Trip Setpoint in TS Table 3.3-4, Engineered Safety Feature Actuation System Instrumentation Trip Setpoints. The change to Table 2.2-1 will lower the Trip Setpoint for Functional Unit 20.C, Power Range Neutron Flux, P-8 from 35 percent to 30 percent of rated power. The first change to Trip Setpoint in TS Table 3.3-4 will increase the Trip Setpoint for Functional Unit 5.a, Turbine Trip and Feed Water Isolation from Steam Generator Water Level -- High-High, from 78.5 percent to 78.9 percent of narrow range instrument span. The second change to Trip Setpoint in TS Table 3.3-4 will increase the Trip Setpoint for Functional Unit 8.b, Engineering Safety Feature Actuation System Interlocks for Low-Low Tavg, P-12, from 544°F to 545°F.

SNC provided justifications for these changes in its submittal dated September 22, 1997, which employed uncertainty calculations using NRC previously approved methodology in WCAP-13751, "Westinghouse Setpoint Methodology for Protection Systems, Southern Nuclear

Operating Company, Farley Nuclear Plant, Units 1 and 2, June 1993," and NSD-NT-OPL-96-152, Revision 2, "Joseph M. Farley Nuclear Plant Units 1 & 2 Licensing Report for Technical Specification Changes Associated with Revised Core Limits, Revised OTΔT/OPΔT Trip Setpoints and Inclusion of RAOC Control Strategy, May 1996."

SNC proposed changes to the Allowable Values in TS Table 2.2-1 and TS Table 3.3-4 (Attachment 1, Tables 4.14-1 and 4.14-2).

In a conference call on September 29, 1997, SNC stated that they have calculations based on the above mentioned approved setpoint methodology covering all of the above Allowable Value changes and that when compared to the current TS values, the proposed changes are conservative.

The staff has reviewed SNC's evaluation and justification for Trip Setpoint and Allowable Value changes in TS Tables 2.2-1 and 3.3-4. SNC stated that the proposed changes have been supported by uncertainty calculations in accordance with NRC-approved methodology in Westinghouse Reports WCAP-13751 and NSD-NT-OPL-96-152 and meet all plant acceptance criteria. The NRC-approved methodology was determined to be applicable to Farley as addressed in Farley License Amendment Nos. 104 and 97, which were issued on December 28, 1993. Based on this review, the staff concludes that the proposed TS changes do not involve a significant reduction in safety margin or a significant increase in the likelihood of a false trip, nor a failure to trip upon demand, and are therefore, acceptable.

4.16 Reactor Trip Time Delays

There are various instrumentation delays associated with each reactor trip function, which are modeled directly and considered in the non-LOCA safety analyses. The total delay time is defined as the time delay from the trip conditions are reached to the time the rods are free to fall. SNC performed an analysis and evaluation to demonstrate that the applicable safety analysis acceptance criteria have been satisfied at the uprate conditions.

The staff has reviewed the proposed changes to reactor trip time delays for the following reactor trip functions:

Item	Reactor Trip Function	Time Delay in Seconds	
		Current	Updated
1.	Overtemperature ΔT	8	12
2.	Overpower ΔT	8	12
3.	High Pressurizer Pressure	2	1

In a conference call on December 9, 1997, SNC confirmed that the increase to 12-second time delay in items 1 and 2 are due to:

1. Filter compensation time of 6 seconds (increased from zero second) as reported in Table 1 of NSD-NT-OPL-96-158, Revision 2, dated May 1996. The staff has previously reviewed this report and found it acceptable by letter dated September 3, 1996.

2. Total Resistance Temperature Detector (RTD) response time of 6 seconds including RTD Bypass Piping and Thermal Lag time of 2 seconds, as indicated in Section 2.3 of the staff's safety evaluation, dated March 11, 1992, related to Amendment Nos. 85 and 92.

In a letter dated November 19, 1997, SNC stated that the response time of Item 3 was based on plant response time testing. Further, in a conference call on December 9, 1997, SNC confirmed that its plant maintenance data from 1978 onwards demonstrated a maximum time delay of 740 milliseconds for the RTDs.

The staff has reviewed the proposed changes and SNC's justification and has determined the subject instruments will meet the new response time requirements. The specific instrument response times were relocated from the Technical Specifications to the Final Safety Analysis Report in Farley License Amendment No. 116 for Unit No. 1 and License Amendment No. 108 for Unit No. 2 issued on September 28, 1995.

5.0 BALANCE OF PLANT EVALUATION

5.1 Main Steam System

SNC performed evaluations of the effects resulting from plant operations at the proposed uprated power level on the main steam system including the main steam isolation valves, SG power operated atmospheric relief valves, main steam bypass valves, and main steam safety valves. Main steam piping thermal expansion at the uprated condition is bounded by the current design basis evaluation. Plant operations at the proposed uprated power level will increase the steam mass flow by approximately 5.3 percent. The above existing components are adequately sized for the uprating. This increase has no significant impact on the main steam piping, valves, in-line components, interconnected branch line, or pipe supports as a result of steam hammer dynamic loads resulting from turbine stop valve closure. Therefore, SNC concluded that plant operations at the proposed uprated power level will have an insignificant or no impact on the main steam system and its associated components.

Based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants, the staff finds that FNP operations at the proposed uprated power level will have an insignificant or no impact on the main steam system and its associated components.

5.2 Steam Dump System

SNC evaluated the steam dump system for the plant operations at proposed uprated power level and concluded it will have an insignificant or no impact on the steam dump system. Based on the experience gained from its review of power uprate applications for similar PWR plants, the staff concludes that operation of the steam dump system at the proposed uprated power level is acceptable.

5.3 Condensate and Feedwater Systems

SNC evaluated the condensate and feedwater systems for the plant operations at 2775 MWt reactor power level and stated that approximately 6 percent increase in condensate flow is necessary to provide sufficient flow to the SGs to support normal operation at power uprate conditions. Therefore, it is necessary to modify the condensate pumps. Since these systems do not perform any safety-related function and their failure will not affect the performance of any safety-related system or component, the staff has not reviewed the impact of plant operations at the proposed uprated power level on the design and performance of these systems.

5.4 Circulating Water System

The circulating water system is designed to remove the heat rejected to the condenser by turbine exhaust and other exhausts over the full range of operating loads, thereby maintaining adequately low condenser pressure. SNC stated that performance of this system was evaluated for power uprate and determined that the system is adequate for uprated power level operation.

Since the circulating water system does not perform any safety function and its' failure will not affect the performance of any safety-related system or component, the staff has not reviewed the impact of plant operations at the proposed uprated power level on the designs and performances of this system.

5.5 Service Water System

The service water system (SWS) is designed to supply cooling water to various nonsafety-related components and heat exchangers in the turbine, reactor, and radwaste buildings during normal plant operation, and to supply cooling water to safety-related systems and other essential equipment during a station blackout event and following a LOCA or main steamline break accident. The SWS heat loads resulting from plant operations at the proposed uprated power level will increase slightly. SNC performed evaluations of the effects of these increases in heat loads on SWS and stated that the SWS has the capacity to accommodate the additional heat loads.

Based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants, the staff finds that plant operations at the proposed uprated power level do not change the design aspects and operations of the SWS and have an insignificant or no impact on the SWS. Therefore, the staff concludes that the FNP SWS is acceptable for operations at the proposed uprated power level.

5.6 Main Turbine Auxiliary Systems

The turbine auxiliary system together with circulating water system and, main and auxiliary condensers, are designed to remove the heat rejected to the condenser by turbine exhaust and other exhausts over the full range of operating loads, thereby maintaining adequately low condenser pressure. SNC stated that performance of these systems were evaluated for power uprate and determined that these systems are adequate for uprated power level operation.

Since these systems do not perform any safety function and their failure will not affect the performance of any safety-related system or component, the staff has not reviewed the impact of the uprated power level operation on the designs and performances of these systems.

5.7 Main Generator and Auxiliaries

SNC has reviewed the potential impact of a power uprate on the main generator stator and rotor, exciter and voltage regulator, hydrogen cooling system, and generator protective relays. Since the power uprate will generate additional heat in the main generator stator and rotor, and in the exciter, the main generator cooling hydrogen pressure must be increased from 72 psig to 75 psig. SNC has reviewed the maximum heat load capability of the hydrogen cooler and maximum exciter cooler heat duty. SNC's review finds that the power uprate does not increase the maximum heat load of the hydrogen cooler beyond its design limits, and the exciter cooler design has sufficient margin to accommodate the heat increase while maintaining the voltage regulation acceptable. Because all protective relay settings are based on the maximum capability of the generator output, SNC finds that the protective relay settings of the main generator did not require any changes for the power uprate conditions.

Based on the foregoing, the staff finds that the proposed power uprate does not necessitate any changes to the existing hydrogen cooling system, exciter cooling system, and generator protective relays.

5.8 Auxiliary Building and Turbine Building Heating, Ventilating, Air Conditioning (HVAC), and Filtration Systems

The HVAC and filtration systems for the following areas were evaluated to ensure that they are capable of supporting the plant uprate conditions:

- Control room and technical support center
- Fuel handling building
- Engineered safety feature (ESF) pump rooms
- Main steam valve rooms
- Turbine driven auxiliary feedwater (AFW) pump room
- Turbine building

5.8.1 Control Room And Technical Support Center (TSC)

The HVAC and filtration systems for the control room and TSC are designed to provide a reliable and suitable environment for plant personnel and equipment under normal and emergency conditions. SNC performed evaluations and concluded that: (1) the post-accident doses meet GDC-19, and (2) safety-related components do not exceed any design limit considering a single active or passive failure HVAC system at the proposed uprated power level.

Based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level do not change the design aspects and operations of the HVAC and filtration systems for the control room and TSC. Therefore, the staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on the HVAC and filtration systems for the control room and TSC.

5.8.2 Fuel Handling Building

Fuel handling building heating, ventilating (HV), and filtration systems are designed to provide adequate capacity to ensure that proper temperatures are maintained during normal, shutdown, and emergency conditions, and to ensure that effluent discharges are maintained within acceptable levels. SNC performed analyses and concluded that the HV and filtration systems for these areas are not impacted by plant operations at the proposed uprated power level.

Based on the staff's review and the experience gained from its review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level do not change the design aspects and operations of the HV and filtration systems in the fuel handling building. Therefore, the staff finds that plant operations at the proposed uprated power level will have an insignificant or no impact on these HV and filtration systems.

5.8.3 ESF Pump Rooms, Main Steam Valve Rooms and Turbine Driven AFW Pump Room

The HV systems for these areas are designed to provide a suitable environment for equipment and personnel. SNC performed evaluations and concluded that the HV systems for these areas are not impacted by plant operations at the proposed uprated power level.

Based on the staff's review and the experience gained from our review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level do not change the design aspects and operations of the HV systems in the fuel handling building. Therefore, the staff finds that plant operations at the proposed uprated power level will have an insignificant or no impact on these HV systems.

5.8.4 Turbine Building

SNC stated that the HVAC systems were evaluated against current performance with respect to potential increased demand on each subsystem to perform the design functions. It was determined that changes to current operation heat loads are negligible.

Since these systems do not perform any safety function and their failure will not affect the performance of any safety-related system or component, the staff has not reviewed the impact of the uprated power level operation on the designs and performances of these systems.

5.9 Radwaste Systems (Solid, Liquid, and Gaseous)

The solid, liquid, and gaseous radwaste activity is influenced by the reactor coolant activity which is a function of the reactor core power. SNC performed evaluations of the existing design of the radwaste systems and concluded that plant operations at the proposed uprated power level will have an insignificant impact on the radwaste systems.

Based on its review, the staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the radwaste systems.

5.10 Additional Balance of Plant Reviews

The impact of plant operations at the proposed power uprate power level on High Energy Line Break Outside Containment and Equipment Environmental Qualification is addressed in sections 4.12 and 4.13.

5.11 Miscellaneous Electrical Reviews

A power uprate that increases the electrical output of the main generator from 885 MWe to 910 MWe, requires increased horsepower for the reactor coolant pumps (RCPs) and condensate pumps (CPs). To determine how those increased non-Class1E loads would affect the station auxiliary electrical distribution system, SNC evaluated the onsite electrical distribution system by using a peak value of 920 MWe, instead of 910 MWe (to be conservative), to ensure that all station auxiliary loads can continue to perform their intended safety-related and nonsafety-related functions. The following electrical systems and components were reviewed:

- main power transformer (MPT)
- startup transformer (SAT)
- unit auxiliary transformer (UAT)
- isophase bus
- station blackout (SBO)
- emergency diesel generators (EDGs)
- CP motors
- RCP motors
- station service assessment
- grid stability

SNC finds that:

1) a comparison of electrical loadings on MPT, SAT, and UAT transformers before and after power uprate, indicates that the transformers would have sufficient capacity and capability to handle the uprated generator output. In addition, the evaluation of those transformer protective relay setpoints for the increased generation during the assumed worst grid voltage conditions indicates that the transformers could reach their present alarm setpoints without exceeding the transformer temperature design limits.

2) the isophase bus for Units 1 and 2 remains capable of supporting the output of the main generator at uprated conditions without modifications since the maximum current flows under power uprate condition are less than those of the rating for the isophase bus. The isophase cooling units are capable of removing the additional heat generated by the

power uprate. No known hot spots currently exist, and none are expected to result from the power uprate condition.

3) the EDG load during the loss of offsite power (LOOP), LOOP with LOCA, and SBO events is within the design ratings of the EDG; thus, the existing EDG loading analysis remains bounding and valid to perform its intended safety-related function. Also, the response time for EDG starting and loading for the above events is not affected by power uprate.

4) the changes to load, voltage, and short circuit values from power uprate conditions at all electrical bus levels are minimally impacted when compared with current values. Total projected loads on the 4160 V buses remain within the rated capacity of the buses, breakers, and transformers. Non-Class1E loads (i.e., RCPs and CPs) that increase as a result of the power uprate are within the design capability of the motors and the cable ratings. As an example, steady state voltages at the 4160 V buses decrease by no more than 0.4 percent, motor starting voltages remain within acceptable limits, and short circuit currents are within acceptable values. Since those load changes were not sufficient to affect the voltage setpoints for the loss of voltage and degraded grid voltage protection relays, no TS revision is needed for the electrical power system.

5) the acceleration time for pumps, motor structural and torque loadings, motor insulation life, and existing relay setpoints are acceptable for operation for the increased horsepower required to support the condensate flow increase. In addition, bus and transformer relay settings are not affected, and also protective relays are adequate for RCP motor protection under uprate conditions.

6) SNC's evaluation of the electrical grid stability impact of increasing FNP generation to 920 MWe indicates that the grid remains stable and that safety-related buses continue to be supplied by the preferred offsite power source for single contingency events and faults.

Upon the staff's request for a sample of onsite load flow studies and grid stability cases, SNC, by letter dated September 22, 1977, submitted one-line diagrams that illustrate the load flows and voltages for before and after power uprate cases at their worst (i.e., minimum) expected grid voltage at Unit 1 (Unit 2 would be the same) and the summary of grid stability studies SNC performed. The staff concurred with SNC that the bus loadings and voltage changes throughout the onsite and offsite power system for the power uprate are minimal at FNP. The result of stability cases summarized under "Single Event Cases Used to Validate FSAR Commitments" reaffirms that the grid will remain stable and that safety-related buses will continue to be supplied by the offsite power source for single contingency events and faults.

On the basis of the above evaluation, for all the secondary-side systems reviewed, the staff finds that the power uprate will have no significant impact on the Balance of Plant design bases.

6.0 HUMAN FACTORS

The staff reviewed SNC's submittals dated February 14 and September 22, 1997, for power uprate. The staff's evaluation of SNC's responses to five review topics is provided below.

Topic 1 - Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will it require any new operator actions?

By letter dated September 22, 1997, SNC stated that the power uprate would not change the type and scope of plant emergency and abnormal operating procedures. SNC also stated that the power uprate would not change the type, scope, or nature of operator actions needed for accident mitigation and that it would not require any new operator actions, with one possible exception. This exception arises as a result of an assumed increase in allowable charging and safety injection pump head degradation allowance from 8 percent to 10 percent, which may require opening a pressurizer PORV. SNC added that (1) the PORV action is consistent with the generic guidance in the Westinghouse Owners Group Emergency Response Guidelines, and (2) the increase in charging pump degradation is not directly related to the power uprate. The staff finds SNC's responses satisfactory.

Topic 2 - Provide examples of operator actions potentially sensitive to power uprate and address whether the power uprate will have any effect on operator reliability or performance. Identify operator actions that would necessitate reduced response times associated with a power uprate. Please specify the expected response times before the power uprate and the reduced response times. What have simulator observations shown relative to operator response times for operator actions that are potentially sensitive to power uprate? Please state why reduced operator response times are needed. Please state whether the reduced time available to the operator as a result of the power uprate will significantly affect the operator's ability to complete manual actions in the times required.

SNC's letter of September 22, 1997, stated that there were no changes made to operator action assumptions in the FSAR Chapter 15 accidents and transients that resulted in reduced operator response times. SNC stated that emergency response procedure operator actions potentially sensitive to power uprate are those that are performed on the basis of setpoint values that are calculated using the design parameters for power uprate. SNC noted that changes in design parameters can affect the setpoints calculated for operator actions but should not affect the type and scope of operator actions. Further, SNC stated that emergency response and normal and operating procedure revisions will be incorporated, where appropriate, before implementation of power uprate. The staff finds SNC's responses acceptable since the subject changes are enhancements that are covered under SNC's existing program, and procedures, for changes to plant procedures.

Topic 3 - Discuss any changes the power uprate will have on control room instruments, alarms, and displays. Are zone markings on meters changed (e.g., the normal range, the marginal range, and the out-of-tolerance range)?

SNC stated in its letter of September 22, 1997, that preliminary engineering reviews indicate that power uprate will have a minimum impact on the control room controls, alarms, and displays. SNC noted several examples of potential control room changes. One potential change concerned color-coding indicators of normal operating steamline pressure, which may be adjusted to a low value of 770 psig. A second potential change involved the setpoint for the reactor coolant system high T-average annunciator. A third potential change dealt with the setpoints for the low suction pressure of the steam generator feed pump. SNC stated that required changes would be implemented. The staff finds SNC's responses satisfactory since the potential changes are enhancements to control room controls, alarms, and displays.

Topic 4 - Discuss any changes the power uprate will have on the Safety Parameter Display System (SPDS).

By letter dated September 22, 1997, SNC stated that on the basis of an SPDS computer point list review, no SPDS setpoint changes are anticipated at this time. SNC explained, however, that there is a potential for changes in the plant process computer and/or SPDS scaling/calibration curves and the high/low alarm limits for some uprate-affected instrumentation inputs that are non-SPDS computer points. The staff finds SNC's responses satisfactory.

Topic 5 - Describe any changes the power uprate will have on the operator training program and the plant simulator. Provide a copy of the post-modification test report (or test abstracts) to document and support the effectiveness of simulator changes as required by ANSI/ANS 3.5-1985, Section 5.4.1.

Specifically, please propose a license condition and/or commitments that address the following:

(a) Provide classroom and simulator training on the power uprate modification.

SNC's letter of September 22, 1997, stated that classroom and simulator training on the uprate changes for Units 1 and 2 will be provided to operations crews before the Unit 2 startup in the spring of 1998. The staff finds this information acceptable.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be revalidated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing."

Simulator revalidation will include comparison of individual simulated systems and components and simulated integrated plant steady-state and transient performance with reference plant responses using similar startup test procedures.

SNC notes that the simulator is referenced to Unit 1, and therefore, final simulator modifications will be implemented following the Unit 1 uprating. SNC also commits to a temporary simulator modification on the basis of the Unit 2 power uprate, including hardware and software changes. Simulator testing will be based on existing simulator certification tests consistent with the requirements of ANSI/ANS 3.5-1985. Best estimate data derived from applicable design change packages and engineering reports will be used to initially validate simulator modifications. The staff finds this information acceptable.

(c) Complete control room and plant process computer system changes as a result of the power uprate.

Hardware and plant process computer modifications will be temporarily implemented before the Unit 2 startup training. Final, permanent implementation will occur following the reference unit modifications. The staff finds this information acceptable.

(d) Modify training and plant simulator relative to issues and discrepancies identified during the startup testing program.

After final modifications to the reference unit and complete implementation of simulator modifications, the simulator will be further evaluated with respect to actual plant performance data and updated accident analysis data. The results of this final testing will be integrated into the quadrennial certification testing program. The staff finds this information acceptable.

On the basis of the information provided by SNC in response to Topic 5, the staff finds the proposed simulator modifications and associated simulator testing plans, as identified in Items (a) through (d) above, are satisfactory with respect to SNC's commitment to ANSI/ANS 3.5-1985, as endorsed by Regulatory Guide 1.149, Revision 1. On the basis of SNC's commitments relative to training, the staff finds that SNC has proposed satisfactory changes to the operator training program as a result of the power uprate. These commitments, as identified in Items (a) through (d), above, have been incorporated into the Facility Operating Licenses for both units as License Conditions. The change is described in Section 7.0 of this safety evaluation.

The staff concludes that the previously discussed review topics associated with the proposed FNP Units 1 and 2 power uprate have been or will be satisfactorily addressed. The staff further concludes that the power uprate should not adversely affect operator performance or operator reliability.

7.0 EVALUATION OF CHANGES TO FACILITY OPERATING LICENSE (FOL) AND TS

The FOL and TS changes requested by SNC in its power uprate submittal are discussed below.

7.1 Evaluation of Changes to the TS and Operating License

7.1.1 License Condition 2.C.(1), Maximum Power Level

The License Condition 2.C.(1) "Maximum Power Level" is being revised from 2652 MWt to 2775 MWt. SNC has provided the results of its reanalyses or evaluation including LOCA and Non-LOCA transients and accidents, containment response, radiological consequences, NSSS and Balance of Plant (BOP) systems and components to support the operation of FNP Units 1 and 2 at the uprated power level. The staff has reviewed SNC's submittal and concludes that both FNP units can safely operate at a core power of 2775 MWt.

7.1.2 New Appendix C, "Additional Conditions," to Operating License

The staff will impose four new license conditions for each unit, to be located in a new Appendix C of the respective operating license. The new license conditions are needed in order to grant approval of the power uprate license amendments. By letter dated April 17, 1998, SNC proposed the following license conditions for Unit 1: (1) SNC shall complete classroom and simulator training for operations crews prior to Unit 2 entering Mode 2 from the spring 1998 refueling outage; (2) SNC shall complete final simulator modifications in accordance with ANSI/ANS 3.5-1985 and review results of the Cycle 16 startup testing to determine any potential effects on operator training within 2 years after restart from the Unit 1 fall 1998 refueling outage; and (3) SNC shall provide an SGTR radiological consequences analysis that incorporates a flashing fraction, which is appropriate for the Unit 1 design prior to the Unit 1 SG replacement outage in spring 2000.

In addition, by letter dated April 17, 1998, SNC proposed the following license conditions for Unit 2: (1) SNC shall complete classroom and simulator training for operations crews and temporary simulator modifications prior to Unit 2 entering Mode 2 from the spring 1998 refueling outage; (2) SNC shall review the results of the Cycle 13 startup testing to determine any potential effects on operator training and incorporate these changes into licensed operator training prior to Unit 1 startup from the fall 1998 refueling outage; and (3) SNC shall provide an SGTR radiological consequences analysis that incorporates a flashing fraction, which is appropriate for the Unit 2 design prior to the Unit 2 SG replacement outage in spring 2001.

7.1.3 TS 1.25, Definitions: Rated Thermal Power

The definition for rated thermal power is being changed to increase the rated power from 2652 MWt to 2775 MWt. SNC has provided the results of its reanalyses or evaluation including LOCA and Non-LOCA transients and accidents, containment response, radiological consequences, NSSS and BOP systems and components to support the operation of FNP Units 1 and 2 at the uprated power level. The staff has reviewed SNC's submittal and concludes that both FNP units can safely operate at a core power of 2775 MWt.

7.1.4 TS Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints

TS Table 2.2-1 was modified to change the setpoints and allowable values for the trip setpoints. The allowable values were based on the FNP instrumentation and were modified to support the amendment request. All applicable LOCA and non-LOCA transient analyses were performed using the new setpoints with acceptable results. The Power Range, Neutron Flux allowable value for the low setpoint was changed from 26 percent to 25.4 percent, and the high allowable value was changed from 110 percent to 109.4 percent. The Power Range, Neutron Flux, High Positive and Negative rate allowable values were changed from 5.5 percent to 5.4 percent. The Pressurizer Pressure values for the low allowable value was changed from 1855 psi to 1862 psi and the high allowable value was changed from 2395 psig to 2388 psig. The Pressurizer Water Level -- High allowable valve was changed from 93 percent to 92.4 percent. The Loss of Flow allowable value of 88.5 percent of 89,290 gpm was changed to 89.7 percent of 88,100 gpm. The Steam Generator Water Level -- low-low allowable value of 23.3 percent was changed to 24.6 percent. The Reactor Trip System Interlocks for Low Power, Power Range Neutron Flux, and the Reactor Trip Block following a turbine trip were adjusted. Because all applicable LOCA and non-LOCA transient analyses were performed using the new setpoints with acceptable results, the staff finds the new setpoints and allowable values acceptable. In addition, the staff concludes that the proposed TS changes do not involve a significant reduction in safety margin or a significant increase in the likelihood of a false trip, or a failure to trip on demand.

7.1.5 TS Table 3.2-1, Departure From Nucleate Boiling (DNB) Parameters and TS Bases B 3/4 4.2.5

The DNB parameters for average temperature (T_{ave}), RCS pressure and RCS flow were changed from 580.7 °F, 2205 psig, and 267,880 gpm to 580.3 °F, 2209 psig, and 264,200 gpm. Where applicable, the NRC-approved revised thermal design procedure (RTDP) continues to be used in the evaluation of the DNB. A reduced core flow of 86,000 gpm per loop was utilized in the analysis, which corresponds to an indicated flow limit of 87,800 gpm per loop. The steam generator tube plugging was assumed to be 20 percent. The limiting DNB events were analyzed using these new values in the safety analysis with acceptable results to assure the new DNB parameters are acceptable. As a result, these changes are acceptable. The TS Bases were also modified to reflect these changes.

7.1.6 TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints

TS Table 3.3-4, Engineered Safety Features Actuation System (ESFAS) Instrumentation Trip Setpoints, was modified to change the setpoints and allowable values for the ESFAS setpoints. The allowable values, based on the FNP instrumentation, were modified to support the amendment request. All applicable LOCA and non-LOCA transient analysis were performed using the new setpoints with acceptable results. The SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION Pressurizer Pressure—Low allowable value was changed, the Steamline ISOLATION high steam flow coincident with low-low average temperature allowable values was modified, the TURBINE TRIP AND FEEDWATER ISOLATION steam generator high-high water level setpoint and allowable value was modified, the AUXILIARY FEEDWATER steam generator low-low level allowable value was modified, and the ESF ACTUATION INTERLOCKS pressurizer pressure and low-low average temperature setpoints and allowable values was modified. The staff finds the new setpoints and allowable values acceptable. In addition, the staff concludes that the proposed TS changes do not involve a significant reduction in safety margin or a significant increase in the likelihood of a false trip, or a failure to trip on demand.

7.1.7 TS 3/4.4.6, SG F* Criteria (Unit 2 Only)

SNC proposed revising the existing F* distance from 1.54 inches to 1.6 inches in the FNP Unit 2 TS. Unit 1 has no F* criteria in the TS. The F* distance is the length of the expanded part of the tube inside the tubesheet that provides a sufficient length of undegraded tube expansion to resist the pullout of the tube from the tubesheet. Under the power uprate, the F* distance needs to be lengthened to provide additional tube friction surfaces to resist the increase of the pullout force resulting from increased primary-to-secondary differential pressure during postulated accidents. The F* criteria in Unit 2 TS 3/4.4.6 are affected by the increased differential pressure between the primary and secondary loops under the power uprate. Based on the staff's previous review, the staff determined that the proposed F* distance of 1.6 inches is acceptable and may be incorporated into the FNP Unit 2 TS.

7.1.8 TS 4.5.2.g. Emergency Core Cooling Systems

The required differential pressure developed by the centrifugal charging pumps and the residual heat removal pumps has been revised from 2458 psig to 2323 psid and 136 psig to 145 psid, respectively. The applicable analysis, including the LOCA and non-LOCA transients, have been reperformed using the reduced flows with acceptable results. The LOCA analysis concluded that limits established in 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, are met. As a result, the change is acceptable.

7.1.9 TS Bases B 3/4.6.1.4, B 3/4.6.1.6, and TS 6.16

SNC proposed to revise the Containment Systems Bases B 3/4.6.1.4 and B 3/4.6.1.6 to reflect the new peak calculated containment pressure obtained from a LOCA event from 48 psig to 43 psig and add that the maximum pressure from an MSLB event is 53 psig. SNC also proposed to indicate that these containment analyses calculations include an initial positive pressure of up to 3 psig and that the analyses results demonstrate that the maximum containment pressure will remain below the design limit of 54 psig. Structural integrity is required to ensure that the containment will withstand the maximum peak calculated internal pressure of 53 psig in the event of an MSLB, including an initial positive pressure of up to 3 psig. In accordance with Appendix J, Option B, the LOCA peak calculated containment internal pressure defines the Pa value for the containment leakage rate testing program required by Surveillance Requirement 4.6.1.2.

In TS Section 6.16, "Containment Leakage Rate Testing Program," SNC proposed to revise the peak calculated containment internal pressure for the design basis LOCA, Pa value from 48 psig to 43 psig.

The revised Bases and test program reflect the results of new containment analyses performed for uprate. The staff has reviewed SNC's submittals and finds the proposed changes acceptable.

7.1.10 TS 4.7.1.5 and TS Bases B 3/4 7.1.1

The required closure time for the main steam isolation valves has been increased from 5 seconds to 7 seconds. The analysis of the main steamline failure was reperformed using the higher time with acceptable results. The analysis of the steamline failure concluded that no DNB would occur, no fuel failure is predicted, and the calculated offsite and onsite doses are acceptable. As a result, the change is acceptable. The TS Bases section for the secondary safety valve was modified to reflect changes to the secondary code safety relieving capacity and total steam flow. Although the total capacity of the secondary code safety relief valves is not being changed, the relative capacity has gone down. The new steam flow in the secondary system is increased from 11,613,849 lbs/hr to 12,270,000 lbs/hr and the relieving capacity is reduced from 112 percent of total flow to 105.8 percent total flow. SNC performed an analysis of overpressure protection and the limiting overpressure transients to verify that there continues to be sufficient secondary relieving capacity. The staff has reviewed this change and concludes that operation at the proposed uprated power level will have an insignificant impact on the main steam system and its associated components. As a result, the changes are acceptable.

7.1.11 TS Table 5.7-1

SNC proposed to increase the number of secondary system hydrostatic tests limit to 10 to allow a relaxation of the current limit of 5 specified in Table 5.7-1,"Component Cyclic or Transient Limits" of the FNP TSs. SNC indicated that the power uprate fatigue analysis assumed a maximum number of occurrences of 10 for the secondary site hydrostatic tests, as shown in a table on page 52 of SNC's August 5, 1997, response to the staff's request for additional information. The staff has reviewed SNC's power uprate evaluation and finds that the proposed change is acceptable.

7.1.12 TS 5.4, Design Features

The value given for the total water volume and steam volume in the RCS is being changed from 9723 to 9829 ft^3. The new value reflects the current calculated total and steam volume of the RCS at 567.2°F. The new calculated RCS fluid volume is supported by the overall uprate program that included the reanalysis or evaluation of the LOCA, non-LOCA, thermal-hydraulic, and nuclear aspects of the NSSS and BOP structures, systems, and components. As a result, the staff finds this acceptable.

7.1.13 TS 6.9, Administrative Controls

The references to WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse Evaluation Model Using BASH Code," is being deleted and replaced with a reference to the approved version of WCAP-12945-P-A, "Code Qualification Document for Best Estimate LOCA Analysis" (W-Proprietary), March 1998. Because the new code has been reviewed and approved for LOCA analysis and the code is used in accordance with all limitations and restrictions, this is acceptable. A reference to the Westinghouse fuel design Topical Report, WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Report," April 1995 (W-Proprietary), using ZIRLO cladding is also being added to the TS. This is acceptable because ZIRLO clad fuel is being used at FNP and the referenced topical report is approved for FNP. The use of these methodologies will ensure that values for cycle-specific parameters are determined such that all applicable limits of the plant's safety analysis are met.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendments. The State official had no comments.

9.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on April 23, 1998 (63 FR 20221).

Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of the amendments will not have a significant effect on the quality of the human environment.

10.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachments: 1. Tables
2. References

Principal Contributors: C. Jackson, J. Wu, J. Tsao, G. West, P. Kang, S. Mazumdar, M. Banic, E. Murphy, J. Hayes, J. Medoff, R. Goel, F. Collins, D. Shum, J. Zimmerman, K. Parczewski

Date: April 29, 1998

TABLES

Attachment 1

Table 3.5.1-1 Assumptions for Loss of AC Power Event

Parameter	Value
Core Thermal Power Level (MWt)	2831
Mass of steam released from three steam generators (lbs)	
(0-2 hours)	448,000
(2-8 hours)	861,000
Feedwater flow to the three steam generators (lbs)	
(0-2 hours)	603,000
(2-8 hours)	953,000
Primary-to-Secondary Leak Rate (gpd/SG)	150
Reactor Coolant concentrations	
Noble Gases	Based upon 1% failed fuel
Dose Equivalent ^{131}I (μCi/g)	60
Secondary Side Activity	
Dose Equivalent ^{131}I (μCi/g)	0.1
Iodine Partition Factor in SGs	0.1
Duration of Plant Cooldown (hrs)	8

Table 3.5.1.3-1 Assumptions for LOCA Analysis

Parameter	Value
Core Thermal Power (MWt)	2831
Activity Released to the Reactor Building	
Airborne (fraction of core)	
Iodine	0.5
Noble Gases	1.0
Elemental Iodine Plateout Rate (1/hr)	
DF ≤ 100	2.7
100 < DF ≤ 1000	0.27
1000 < DF	0
Iodine Species (fraction)	
Elemental	0.955
Particulate	0.025
Organic	0.02
Activity Released to Sump (fraction)	
Iodine	0.5
Noble Gases	0.0
Containment Free Volume (ft^3)	2.03E6
Leakage Rate (percent/day)	
0-24 hours	0.15
> 24 hours	0.075

Table 3.5.1.3-1 Assumptions for LOCA Analysis (cont.)

Parameter	Value
Sump Liquid Volume (ft^3)	4.92E4
Reactor Building Spray System	
Actuation Time (sec)	30
Spray Removal Constants (/hr)	
Elemental	10
Particulate	
DF $\leq$ 50	4.8
50 < DF	0.48
Fraction of Reactor Building Unsprayed	0.178
Recirculation Loop Leakage Rate (cc/hr)	7.78E4
Minimum Time to Recirculation (min)	20
Passive Component Failure Leak Rate (gpm) for 30 minutes @24 hours post-LOCA	NA
Control Room Free Volume (ft^3)	1.16E5
Filtered Recirculation Flow (cfm)	2700
Control Room Recirculation Filtration System Efficiency for all forms of Iodine (%)	95
Control Room Makeup Air Filtration Rate (cfm)	270
Control Room Makeup Filter Efficiency for all forms of Iodine (%)	99
Control Room Unfiltered Air Infiltration Rate (cfm)	10

Table 3.5.1.3-1 Assumptions for LOCA Analysis (cont.)

Parameter	Value
Control Room Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4
Atmospheric Dispersion Factors (sec/m^3)	
EAB	6.5E-4
LPZ	
0-8 hours	1.0E-4
8-24 hours	6.6E-5
1-4 days	2.6E-5
4-30 days	6.8E-6
Control Room	
0-8 hours	3.3E-3
8-24 hours	2.2E-3
1-4 days	1.6E-4
4-30 days	1.1E-4
Breathing Rates (m^3/sec)	
Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4

Table 3.5.1.5-1 Assumptions for SGTR Accident

Parameter	Value
Iodine Partition Factor	0.1
Steam Release from Defective Steam Generator	
0-0.5 hours(lbs)	7.33E4
0.5-8 hours (lbs)	0
Steam Release from Intact SGs (lbs)	
0-2 hours	3.39E5
2-8 hours	7.30E5
Feedwater Flow to Intact SGs (lbs)	
0-2 hours	4.42E5
2-8 hours	7.91E5
Reactor Coolant Released to Faulted Steam Generator (lbs)	1.50E5
Primary-to-Secondary Leak Rate for Intact Steam Generator (gpd/SG)	150
Isolate Faulted SG (min)	30
Primary Coolant Activity Level - Dose Equivalent ^{131}I (μCi/g)	
Pre-existing Spike	60
Accident Initiated Spike	1
Flashing Fraction	
0-20 minutes	0.18
20-28.5 minutes	0.25
28-60 minutes	0.10

Table 3.5.1.6-1 Assumptions for Locked Rotor Accident

Parameter	Value
Core Thermal Power Level (MWt)	2831
Duration of Plant Cooldown by Secondary System (hr)	8
Gap Fraction:	
^{131}I	0.12
^{85}Kr	0.30
All others	0.10
Failed Fuel Rods (%)	20
Primary-to-Secondary Leak Rate (gpd/SG)	150
Iodine Partition Factor in Steam Generators	0.01
Steam Released from three SGs (lbs)	
0-2 hours	4.27E5
2-8 hours	8.20E5
Feedwater Delivered to three SGs (lbs)	
0-2 hours	5.74E5
2-8 hours	9.08E5

Table 3.5.1.7-1 Assumptions for Fuel Handling Accidents

Parameter	Value
Core Power (MWt)	2831
Total Number of Assemblies in Core	157
Highest Power Discharged Assembly	
Axial Peak to Average Ratio	1.7
Radial Peak to Average Ratio	1.7
Occurrence of Accident (hours after shutdown)	100
Damaged Fuel Rods	264
Fractional Activity Released from the Gap	
All isotopes except as noted below	0.10
^{85}Kr	0.30
^{131}I	0.12
Iodine Gap Inventory	
organic(%)	0.25
inorganic(%)	99.75
Pool DF	
organic(%)	1
inorganic(%)	133
Purge Isolation Time (seconds)	NA

Table 3.5.1.7-1 Assumptions for Fuel Handling Accidents (cont.)

Parameter	Value
Adsorber Efficiency	
Containment Purge Exhaust Filter System	
organic (%)	30
inorganic (%)	90
Penetration Room Filtration System	
organic (%)	70
inorganic (%)	90

Table 3.5.1.8-1 Assumptions for Rod Ejection Accident

Parameter	Value
Core Thermal Power (MWt)	2831
Fuel Defects	
Clad Failure (%)	10
Fuel Melting (%)	0.25
Primary-to-Secondary Leak Rate (gpd/SG)	150
Percent of Fuel which melts and releases activity to reactor coolant	
Noble Gases (%)	100
Iodines (%)	50
Percent of Fuel which melts and releases activity to containment	
Noble Gases (%)	100
Iodines (%)	25
Iodine Partition Factor in the SGs before and after the accident	0.01
Containment Volume (ft^3)	2.03E6
Containment Leak Rate (%/day)	
t = 0-1 day	0.15
t > 1 day	0.075
Iodine Form in Containment (fraction)	
Particulate	0.05
Organic	0.04
Elemental	0.91
Steam Dump from Relief Valves (lbs)	4.26E5
Duration of Steam Dump from Relief Valves (sec)	98
Time between Accident and Equalization of Primary-to-Secondary System Pressure (sec)	2500

Table 3.5.2-1 Thyroid Doses from Postulated Accidents (Rem)

	Accident	EAB	LPZ
1.	Loss of AC Power	0.32	0.14
2.	Large Break LOCA		
	Containment	200	82
	ECCS Leakage	21	52
	Initial Purge	13	1.9
	H_2 Purge	0	31
3.	Main Steamline Break		
	Coincident Spike	13	29.8
	Pre-existing Spike	24	13
4.	Steam Generator Tube Rupture		
	Coincident Spike	39	6.6
	Pre-existing Spike	270	42
5.	Locked Rotor	1.3	1.9
6.	Fuel Handling Accident		
	Inside Containment	41	6.3
	Spent Fuel Pool Area	25	3.8
7.	Rod Ejection		
	Secondary Side Pathway	2.4	0.37
	Containment Pathway	39	80
8.	Small Break LOCA	83	35

Table 3.5.2-2 Whole Body Doses from Postulated Accidents (Rem)

	Accident	EAB	LPZ
1.	Loss of AC Power	0.016	0.0052
2.	Large Break LOCA		
	Containment	4.2	1.6
	ECCS Leakage	-	-
	Initial Purge	-	-
	H_2 Purge	-	0.077
3.	Main Steamline Break		
	Coincident Spike	<1	<1
	Pre-existing Spike	<1	<1
4.	Steam Generator Tube Rupture		
	Coincident Spike	<1	<1
	Pre-existing Spike	<1	<1
5.	Locked Rotor	0.32	0.10
6.	Fuel Handling Accident		
	Inside Containment	0.39	0.059
	Spent Fuel Pool Area	0.39	0.059
7.	Rod Ejection		
	Secondary Side Pathway	0.086	0.013
	Containment Pathway	0.16	0.084
8.	Small Break LOCA	0.65	0.21

Table 3.5.2-3 Control Room Operator Doses from Postulated Accidents (Rem)

	Accident	Thyroid	Whole Body
1.	Loss of AC Power	0.0204	0.0075
2.	Large Break LOCA		
	Containment	13	2.2
	ECCS Leakage	6.9	-
	Initial Purge	-	-
	H_2 Purge	3.2	0.06
3.	Main Steamline Break		
	Coincident Spike	2.0	<1
	Pre-existing Spike	4.4	<1
4.	Steam Generator Tube Rupture		
	Coincident Spike	0.97	<1
	Pre-existing Spike	6.1	<1
5.	Locked Rotor	0.29	0.15
6.	Fuel Handling Accident		
	Inside Containment	0.92	0.086
	Spent Fuel Pool Area	0.56	0.086
7.	Rod Ejection		
	Secondary Side Pathway	0.54	0.019
	Containment Pathway	8.7	0.12
8.	Small Break LOCA	5.3	0.30

Table 4.15-1, Changes to TS Table 2.2-1 Allowable Values

FUNCTIONAL UNIT	DESCRIPTION	CURRENT VALUE	UPRATE VALUE
2	Low Setpoint Power Range Neutron Flux (less than or equal to)	26%	25.4%
		of Rated Thermal Power	
	High Setpoint Power Range Neutron Flux (less than or equal to)	110%	109.4%
		of Rated Thermal Power	
3	Positive Rate Power Range Neutron Flux (less than or equal to)	5.5%	5.4%
		of Rated Thermal Power	
4	Negative Rate Power Range Neutron Flux (less than or equal to)	5.5%	5.4%
		of Rated Thermal Power	
9	Pressurizer Pressure--Low (greater than or equal to)	1855 psig	1862 psig
10	Pressurizer Pressure--High (less than or equal to)	2395 psig	2388 psig
11	Pressurizer Water Level--High (less than or equal to)	93%	92.4%
		of Instrument span	
12	Loss of Flow (greater than or equal to)	88.5%	89.7%
		of minimum measured flow per loop	
13	Steam Generator Water Level--Low-Low (greater than or equal to)	23.3%	24.6%
		of narrow range instrument span	
20.B	Reactor Trip System Interlocks Low Power Reactor Trips Block, P-7 (less than or equal to)	11%	10.4%
		of Rated Thermal Power	
20.C	Reactor Trip System Interlocks Power Range Neutron Flux, P-8 (less than or equal to)	36%	30.4%
		of Rated Thermal Power	
20.D	Reactor Trip System Interlocks Power Range Neutron Flux, P-10 (greater than or equal to)	7%	7.6%
		of Rated Thermal Power	
20.F	Reactor Trip System Interlocks Reactor Trips Block Following Turbine Trip,P-9 (less than or equal to)	51%	50.4%
		of Rated Thermal Power	

Table 4.15-2, Changes to TS Table 3.3-4 Allowable Values

FUNCTIONAL UNIT	DESCRIPTION	CURRENT VALUE	UPRATE VALUE
1.d	Safety Injection, Turbine Trip and Feedwater Isolation, Pressurizer Pressure--Low (greater than or equal to)	1840 psig	1847 psig
4.d	Steamline Isolation, Steam Flow in Two Steamlines--High, Coincident with Tavg--Low-Low		
	0-20% Load (less than or equal to)	44% of full steam flow	40.3% of full steam flow
	At 100% Load (less than or equal to)	111.5% of full steam flow	110.3% of full steam flow
	T_{avg} (greater than or equal to)	540°F	542.6°F
5.a	Turbine Trip and Feedwater Isolation, Steam Generator Water Level--High-High (less than or equal to)	80.5% of narrow range instrument span	78.9% of narrow range instrument span
6.b	Auxiliary Feedwater, Steam Generator Water Level--Low-Low (greater than or equal to)	23.3% of narrow range instrument span	24.6% of narrow range instrument span
8.a	Engineered Safety Feature Actuation System Interlocks, Pressurizer Pressure, P-11 (less than or equal to)	2010 psig	2003 psig
8.b	Engineered Safety Feature Actuation System Interlocks, Low-Low Tavg, P-12		
	Increasing (less than or equal to)	547°F	545.4°F
	Decreasing (greater than or equal to)	540°F	542.6°F

Table 4.1.1-1. Comparison of NRC and SNC Determined End of Life (EOL) Upper-Shelf Energy (USE) Values for the Beltline Plate and Weld Materials in the FNP Unit 1 Reactor Pressure Vessel [1]

Plate No.	Cu Content (wt. %)	1/4T Fluence (E19 n/cm^2)[2]	Unirradiated USE (ft-lb) [1]	% Decrease in USE [1]	Decrease in USE (ft-lb) [1]	NRC EOL USE @ 1/4T (ft-lb)	SNC Projected 1/4T USE (ft-lb)[1]	
							% Decrease	EOL USE
Int. Shell B6903-2	0.13	2.705	99.0	27.5	27.2	72	28	71
Int. Shell B6903-3	0.12	2.705	87.0	26.3	22.9	64	27	64
Low. Shell B6919-1	0.14	2.705	86.0	23.5 [3]	20.2 [3]	65 [3]	29 [3]	65 [3]
Low. Shell B6919-2	0.14	2.705	86.0	28.8	24.8	61	24	61
Int Shell Axial Welds 19-894 A/B	0.258	0.841	149.0	23.1 [3]	34.4 [3]	115 [3]	23 [3]	115 [3]
Circ Weld 11-894	0.205	2.705	104.0	43.5	45.2	59	43	59
Low. Shell Axial Welds 20-894 A/B	0.197	0.841	82.5	32.5	26.8	56	34	54

Footnotes:

1. All projections for the EOL USE values were performed in accordance with the methodology of Position 1.2 of Section C of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," except as noted in Footnote No. 3.

2. Calculational methods for calculating the neutron fluence at the 1/4 thickness (1/4T) location of the reactor pressure vessel (RPV) wall are as defined in Equation (3) of Regulatory Position 1.1 in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," or in the equivalent equation in the revised rule 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock."

3. Projections for the EOL USE values using surveillance data were performed in accordance with the methodology of Regulatory Position 2.2 of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

Table 4.1.1-2. **Comparison of NRC and SNC Determined End of Life (EOL) USE Values for the Beltline Plate and Weld Materials in the FNP Unit 2 eactor Pressure Vessel [1]**

Plate No.	Cu Content (wt. %)	1/4T Fluence (E19 n/cm²) (2)	Unirradiated USE (ft-lb) (1)	% Decrease in USE (1)	Decrease in USE (ft-lb) [1]	NRC EOL USE @ 1/4T (ft-lb)	SNC Projected USE (ft-lb) (1) % Decrease	SNC Projected USE (ft-lb) (1) EOL USE
Lower Shell B7210-2	0.140	2.736	99.0	28.9	28.6	70	29	70
Int. Shell B7203-1	0.140	2.736	100.0	28.9	25.1	71	29	71
Int. Shell B7212-1	0.200	2.736	100.0	41.4 (3)	41.4 (3)	59 (3)	39 (3)	61 (3)
Lower Shell B7210-1	0.130	2.736	103.0	27.6	28.4	75	28	74
Circ. Weld 11-923	0.153	2.736	102.0	37	37.7	64	37	64
Lower Shell Axial Welds 20-923 A/B	0.057	0.873	126.0	18.5	23.3	103	18.5	103
Int. Shell Axial Weld 19-923 B	0.027	0.873	148.0	6.9 (3)	10.2 (3)	139 (3)	9 (3)	135
Inr. Shell Axial Welds 19-923 A	0.027	0.873	131.0	18.4	24.1	107	18.5	107

Footnotes:

1. All projections for the EOL USE values were performed in accordance with the methodology of Position 1.2 to Section C. of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," except as noted in Footnote No. 3.

2. Calculational methods for calculating the neutron fluence at the 1/4 thickness (1/4T) location of the reactor pressure vessel (RPV) wall are as defined in Equation (3) of Regulatory Position 1.1 in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," or in the equivalent equation in the revised rule 10 CFR 50.61, " Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock."

3. Projections for the EOL USE values using surveillance data were performed in accordance with the methodology of Regulatory 2.2 of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

REFERENCES

Attachment 2

1. February 14, 1997 - Letter from D. Morey, Vice President - Farley Project, Southern Nuclear Operating Company (SNC) to the U.S. Nuclear Regulatory Commission (USNRC) Document Control Desk, "Joseph M. Farley Nuclear Plant Facility Operating Licenses and Technical Specifications Change Request for Power Uprating."

2. June 20, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Facility Operating Licenses and Technical Specifications Change Request for Power Uprating."

3. July 1, 1997 - Letter from J. Zimmerman, Project Manager, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, "Request for Additional Information Related to Power Uprate Submittal - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M98120 and M98121)."

4. July 23, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Technical Specifications Change Request, Pressure Temperature Limits Report."

5. July 24, 1997 - Letter from J. Zimmerman, Project Manager, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, "Supplement to July 1, 1997, Request for Additional Information Related to Power Uprate Submittal - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M98120 and M98121)."

6. August 5, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

7. August 21, 1997 - Letter from J. Zimmerman, Project Manager, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, "Request for Additional Information Related to Power Uprate Submittal - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M98120 and M98121)."

8. September 22, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

9. October 14, 1997 - Letter from J. Zimmerman, Project Manager, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, "Request for Additional Information Related to Power Uprate Submittal - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M98120 and M98121)."

10. November 19, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

11. December 9, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Radiological Analyses."

12. December 17, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

13. December 18, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Technical Specifications Change Request, Pressure Temperature Limits Report."

14. December 31, 1997 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

15. January 23, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

16. February 12, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

17. February 26, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

18. March 3, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

19. March 6, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

20. March 16, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

21. March 31, 1998 - Letter from H. Berkow, Project Director, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, "Acceptance For Referencing of Pressure Temperature Limits Report."

22. Section 50.61 to Part 50 of Title 10, *Code of Federal Regulations*, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock."

23. Appendix G to Part 50 of Title 10, *Code of Federal Regulations*, "Fracture Toughness Requirements."

24. Appendix H to Part 50 of Title 10, *Code of Federal Regulations*, "Reactor Vessel Material Surveillance Program Requirements."

25. January 31, 1996 - Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits."

26. August 3, 1995 - Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

27. August 1976 - "Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

28. Westinghouse WCAP-7588, Revision 1A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Spacial Kinetics Methods," January 1975.

29. Westinghouse WCAP-9227-P-A, Revision 1, "Reactor Core Response to Excessive Secondary Steam Releases," February 1998.

30. Westinghouse WCAP-10054-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Using NOTRUMP Code," August 1985.

31. Westinghouse WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor," January 1983.

32. Westinghouse WCAP-11875, "Joseph Farley Unit 1 and 2, Evaluation For Tube Vibration Induced Fatigue," July 1988. (Proprietary)

33. Westinghouse WCAP-12825, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plants," January 1991. (Proprietary)

34. Westinghouse WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998

35. Westinghouse WCAP-13751, "Westinghouse Setpoint Methodology for Protection Systems, Southern Nuclear Operating Company, Farley Nuclear Plant, Units 1 and 2," June 1993.

36. Westinghouse WCAP-14723, "Farley Nuclear Plant Units 1 and 2, Power Uprate Project NSSS Licensing Report," January 1997. (Nonproprietary)

37. August 12, 1991 - Letter from S.T. Hoffman, Project Manager - USNRC, to W.G. Hairston, III, Senior Vice President, Alabama Power Company, "Safety Evaluation of Elimination of Dynamic Effects of Postulated Primary Loop Pipe Ruptures from Design Basis for Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M79660 and M79661)."

38. March 11, 1992 - Letter from S.T. Hoffman, Project Manager, USNRC, to W.G. Hairston, III, Senior Vice President, Alabama Power Company, Issuance of Amendment No. 92 to Facility Operating License No. NPF-2 and Amendment No. 85 to Facility Operating License No. NPF-8 - Joseph M. Farley Nuclear Plant, Unit 2 (TAC Nos. M81025 and M81026).

39. April 7, 1995 - Letter from B.L. Siegel, Senior Project Manager, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, Issuance of Amendment No. 114 to Facility Operating License No. NPF-2 and Amendment No. 105 to Facility Operating License No. NPF-8 - Joseph M. Farley Nuclear Plant, Unit 2 (TAC Nos. M89839 and M89840).

40. September 3, 1996 - Letter from B.L. Siegel, Senior Project Manager, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, Issuance of Amendment No. 121 to Facility Operating License No. NPF-2 and Amendment No. 113 to Facility Operating License No. NPF-8 - Joseph M. Farley Nuclear Plant, Unit 2 (TAC Nos. M95700 and M95701).

41. October 11, 1996 - Letter from J. Zimmerman, Project Manager, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, Issuance of Amendment No. 115 to Facility Operating License No. NPF-8 - Joseph M. Farley Nuclear Plant, Unit 2 (TAC No. M95146).

42. March 24, 1997 - Letter from J. Zimmerman, Project Manager, USNRC, to D.N. Morey, Vice President - Farley Project, SNC, Issuance of Amendment No. 124 to Facility Operating License No. NPF-2 - Joseph M. Farley Nuclear Plant, Unit 1 (TAC No. M97510).

43. October 1978 - NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels."

44. 1988 - NUREG-0800, "U.S. Nuclear Regulatory Commission Standard Review Plan, Section 6.5.2, Containment Spray as a Fission Product Cleanup System, Revision 2."

45. SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," February 18, 1997.

46. April 3, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Power Uprate Facility Operating Licenses and Technical Specification Change Request."

47. April 13, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

48. April 17, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request. "

49. April 17, 1998 - Letter from D. N. Morey, Vice President - Farley Project, SNC, to the USNRC Document Control Desk, "Joseph M. Farley Nuclear Plant License Conditions Related to Power Uprate Facility Operating Licenses and Technical Specifications Change Request."